SAND89-7002
Unlimited Release
Printed June 1992

Distribution
Category UC-814

PRECLOSURE RADIOLOGICAL SAFETY ANALYSIS FOR THE EXPLORATORY SHAFT FACILITIES

SAND--89-7002

DE92 017360

by

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ABSTRACT

This study assesses which structures, systems, and components of the exploratory shaft facility (ESF) are important to safety when the ESF is converted to become part of the operating waste repository. The assessment follows the methodology required by DOE Procedure AP-6.10Q. Failures of the converted ESF during the preclosure period have been evaluated, along with other underground accidents, to determine the potential offsite radiation doses and associated probabilities. The assessment indicates that failures of the ESF will not result in radiation doses greater than 0.5 rem at the nearest unrestricted area boundary. Furthermore, credible accidents in other underground facilities will not result in radiation doses larger than 0.5 rem, even if any structure, system, or component of the converted ESF fails at the same time. Therefore, no structure, system, or component of the converted ESF is important to safety.

This work was completed January 1989 and was based on the Title I design of the ESF (DOE, 1988).

This report was worked on under the earlier WBS element 1.2.4.6.3, but was completed under the current WBS element 1.2.1.4.3.2.

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EXECUTIVE SUMMARY

This assessment determines which structures, systems, and components of the exploratory shaft facility (ESF) are important to safety. If the site is found suitable during site characterization, the ESF will be converted to become part of the operating waste repository. Because there will be no radioactive waste in the ESF during site characterization, ESF items that will be removed at the end of site characterization cannot be important to safety. Therefore, this assessment evaluates only the items of the ESF that will remain during repository operations.

Most of the ESF items will be removed after site characterization and before repository operations. The remaining ESF items will be the underground openings (excavations and shafts), shaft liners (unreinforced concrete), and ground support (rock bolts and other support features). The study evaluates the potential failures and the associated effects of failures of these remaining ESF items. Only potential accidents that could occur during the preclosure period are addressed in this study (including waste emplacement, caretaker, and decommissioning periods).

The assessment follows the methodology required by DOE Procedure AP-6.10Q (Appendix A) and was based on the Title I design of the ESF (DOE, 1988). External and internal initiating events are identified and screened to determine the events that require further assessment. Seismic events are the only external initiating events surviving the screening criteria. Internal initiating events include spontaneous collapses of drifts or shaft liners. The assessment also considers internal events in other areas of the underground repository (such as transporter collisions and transporter hoist failures) in combination with failures of the converted ESF.

Accident scenarios associated with failures of ESF items are developed and described by event trees. The radiological consequence and the annual probability of occurrence of each of the identified scenarios are determined. On the basis of the probability estimates, specific scenarios in the event trees are determined to be either credible or not credible.

The assessment is based on results of previous preclosure radiological safety analyses of the potential Yucca Mountain repository (Ma, 1988 and MacDougall, 1987).

The major conclusion is that failures of the converted ESF and credible accidents in other underground repository facilities will not result in radiation doses greater than 0.5 rem at the nearest unrestricted area boundary, even if another item in the converted ESF fails at the same time. Credible scenarios that result in offsite radiation doses greater than 0.5 rem are designated as Q-scenarios, which are further evaluated to identify items important to safety in accordance with the DOE procedure in AP-6.10Q. However, no credible scenarios associated with ESF failures result in doses that exceed the 0.5 rem criterion; therefore, no ESF items are important to safety. From this study, no items are identified on the list of items important to safety for the ESF.

This report, including the required list of references and sources of information, comprises the documentation demonstrating that each step of DOE Procedure AP-6.10Q has been completed for the assessment of ESF items important to safety.

1.0 INTRODUCTION

1.1 Purpose

The purpose of this assessment is to determine which structures, systems, or components of the exploratory shaft facility (ESF) are important to safety and was based on the Title I design of the ESF (DOE, 1988). The assessment will contribute to the development of design criteria and quality assurance requirements for the ESF. The results will also be used in the next phase of radiological safety analyses during the Advanced Conceptual Design.

1.2 Scope

This assessment considers only the preclosure period of the potential Yucca Mountain repository. During repository operations, only portions of the ESF will remain, including underground openings (excavations and shafts), shaft liners, and ground support. Potential failures of these items in the converted ESF are evaluated (including failures in combination with other accidents) using the repository configuration identified in the Site Characterization Plan - Conceptual Design Report (SCP-CDR) as a basis (MacDougall, 1987). The assessment follows the method required by DOE Procedure AP-6.10Q (Appendix A). The results of the assessment include (1) a list of ESF items important to safety, (2) a list of ESF items not important to safety, and (3) the report documentation (including a list of references and sources of information) demonstrating that each step of the DOE procedure has been completed.

1.3 Organizational Approach

This report is organized into seven sections, including this introduction. The remaining six sections are as follows:

- Section 2.0, Facility Description, which describes both the ESF and the reference repository configuration containing the converted ESF.

- Section 3.0, Bases for the Assessment, which defines "important to safety" and identifies the method and assumptions used in this assessment

- Section 4.0, Development of Potential Accident Scenarios, which addresses the question of potential accidents in the repository underground facilities associated with the failures of the ESF

- Section 5.0, Event Tree Analyses, which evaluates the consequences and probabilities of accident scenarios

- Section 6.0, Identification of Items Important to Safety, which presents the assessments that result in identification of ESF items important to safety

- Section 7.0, References, which presents a list of references and sources of information.

DOE Procedure AP-6.10Q, Identification of Items Important to Safety, is attached as Appendix A.

2.0 FACILITY DESCRIPTION

2.1 Description of the ESF During Site Characterization

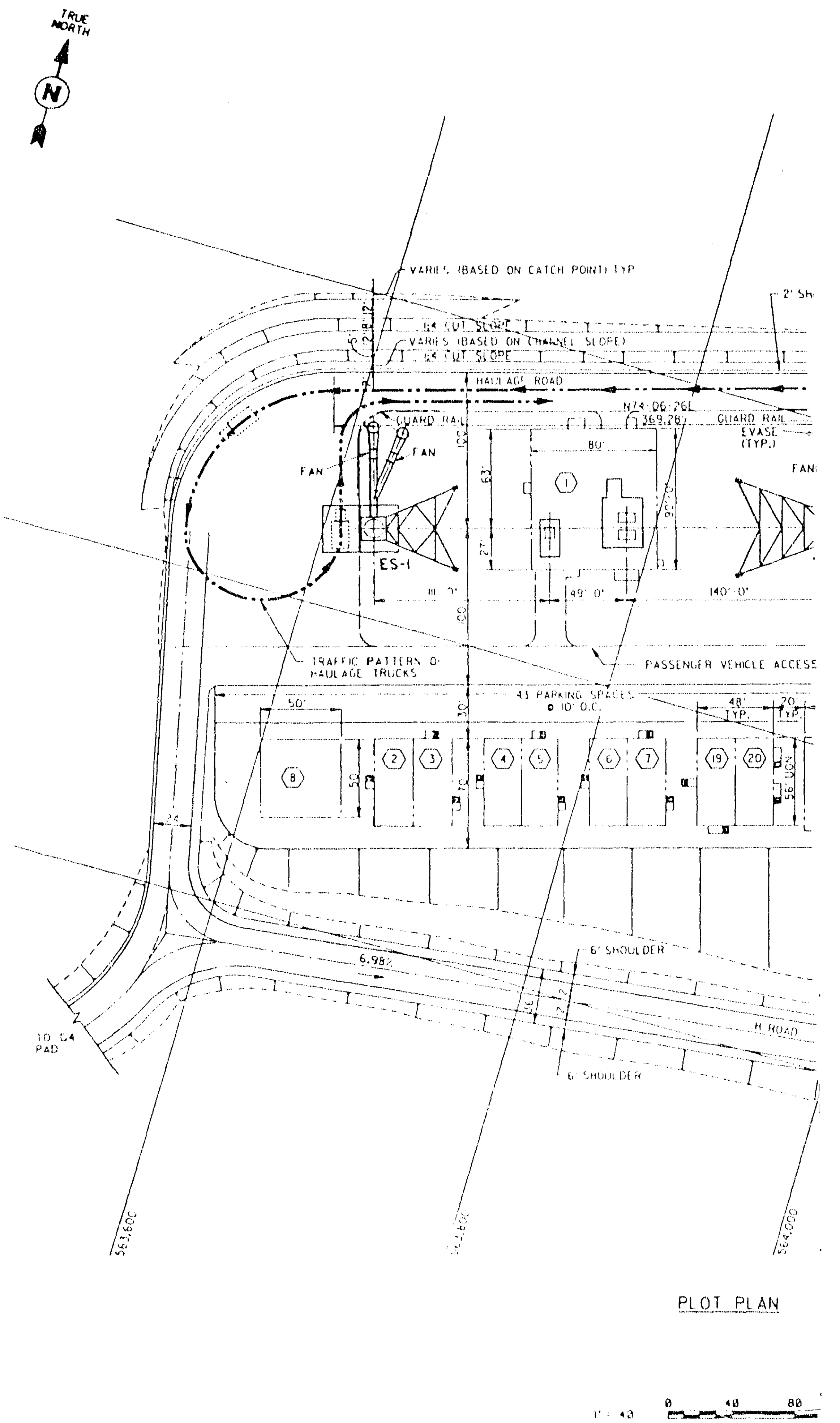
The reference configuration of the ESF used in this report corresponds to the ESF Title I design, as described in DOE (1988). The ESF will be constructed at Coyote Wash on the eastern side of Yucca Mountain approximately halfway up the south side of Dead Yucca Ridge and will consist of support facilities on the surface, two exploratory shafts, and underground testing rooms and drifts. There will be no radioactive materials in the ESF during site characterization (except very small quantities potentially used in measurement systems and equipment that support site characterization).

2.1.1 Surface Facilities

The surface facilities will be constructed on a main pad and several auxiliary pads at the site (DOE, 1988). The main pad contains the shafts, hoist house, and facilities for shaft sinking and operations development. Figure 2-1 shows the layout of the main pad. The auxiliary pads include the booster pump station pad, batch plant and aggregate stockpile pad, possible topsoil storage pad, equipment storage, muck storage pad, substation and compressor pad, explosives storage pad (located far from other pads), G-4 pad (currently existing pad that can be used for storage of supplies), water tank pad, and parking pads.

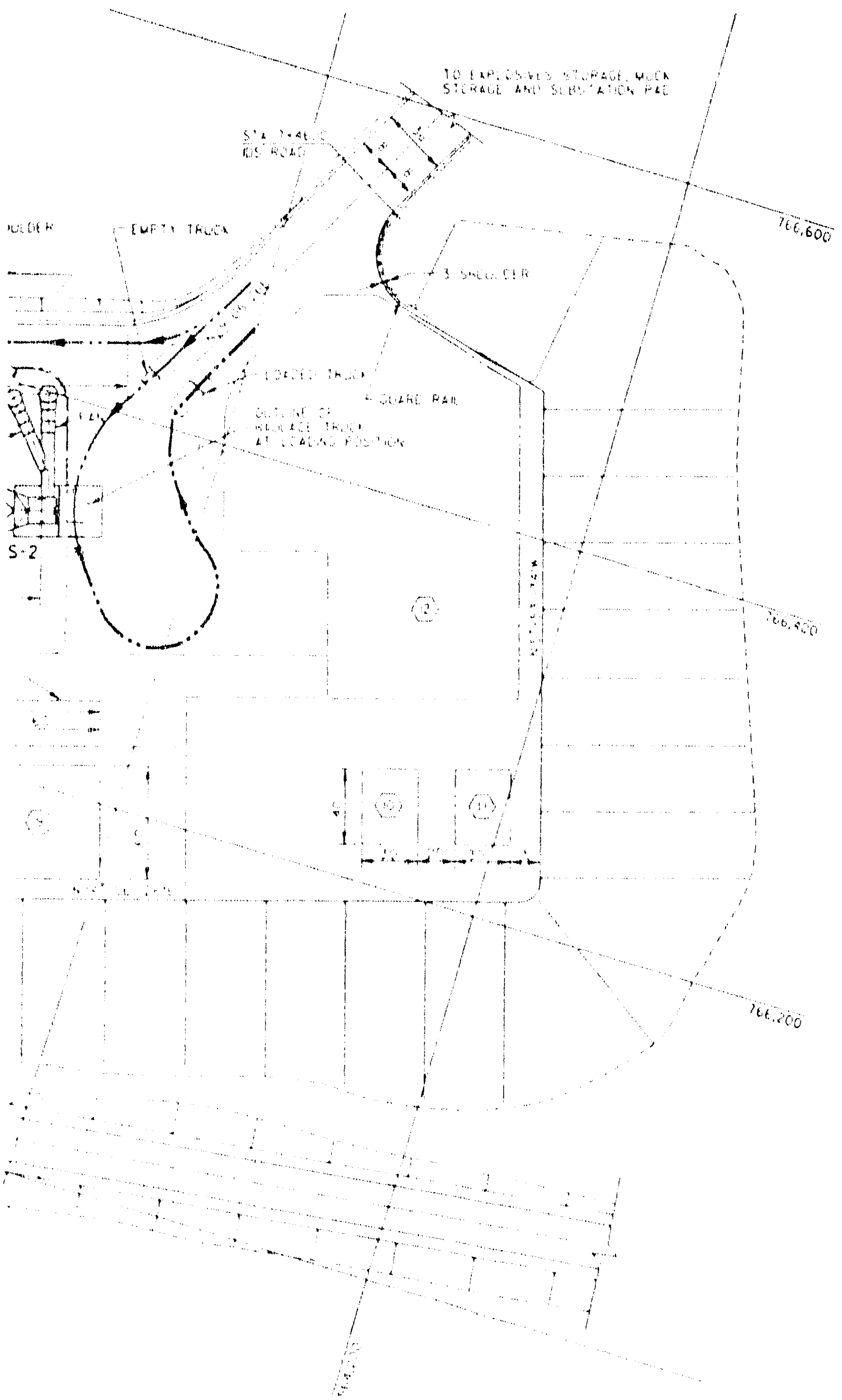
The ESF surface buildings will consist of pre-engineered metal structures on concrete slabs or portable double-wide trailers (DOE, 1988). All utilities are trenched and routed to the required areas. Electrical power will be routed from existing lines at the boundary of the Nevada Test Site. Water will be supplied by an existing well (J-13).

The shaft collars will be in bedrock and will extend from the surface to about 16 ft below the surface. The reinforced-concrete collars provide a foundation for anchoring the headframe.



PLOT PLAN

1" = 40'



LEGEND

1 HOIST HOUSE
2 LANL OFFICE
3 LLNL OFFICE
4 SNL/USGS OFFICE
5 SAIC OFFICE
6 H & N/F & S OFFICE
7 DOE/NRC/STATE OF NEVADA
8 IDS BUILDING
9 MEN'S CHANGE HOUSE/REECO SAFETY
10 SHOP
11 UNASSIGNED
12 SUBCONTRACTORS AREA
19 REECO TRAILER
20 REECO TRAILER

Figure 2-1

Layout of Surface Facilities of the ESF Main Pad
(Source: DOE, 1988)

(Drawing No. FS-GA-0011, Rev. C)

2.1.2 Exploratory Shafts

Two exploratory shafts (ES-1 and ES-2) will be constructed (DOE, 1988). Both shafts will be lined with concrete at least 1 ft thick and will have an inside finished diameter of 12 ft. The shafts will include internal structures, conduits, piping, ventilation ducts, and conveyances to move people and materials.

The shaft liners will be constructed of cast-in-place unreinforced concrete with a compressive strength of 5,000 psi (DOE, 1988). The liners perform the following functions:

- Prevent rockfall hazards

- Serve as a stable and well defined anchorage for the installation and alignment of shaft equipment

- Provide a smooth, low-friction surface for the efficient flow of ventilation air

- Protect the wall rock against weathering

The construction joints between the successive concrete pours will be about 20 ft apart. The joints will not be watertight, and some seepage may occur through the joints if groundwater is present.

Figure 2-2 shows the relationship between exploratory shafts and underground facilities.

2.1.3 Underground Facilities

The first exploratory shaft (ES-1) will provide access to underground facilities at two elevations, one approximately 600 ft below the surface and the other approximately 1,050 ft below the surface (DOE, 1988). ES-1 may also provide access to a station and a drill room approximately 1,360 ft below the surface in the

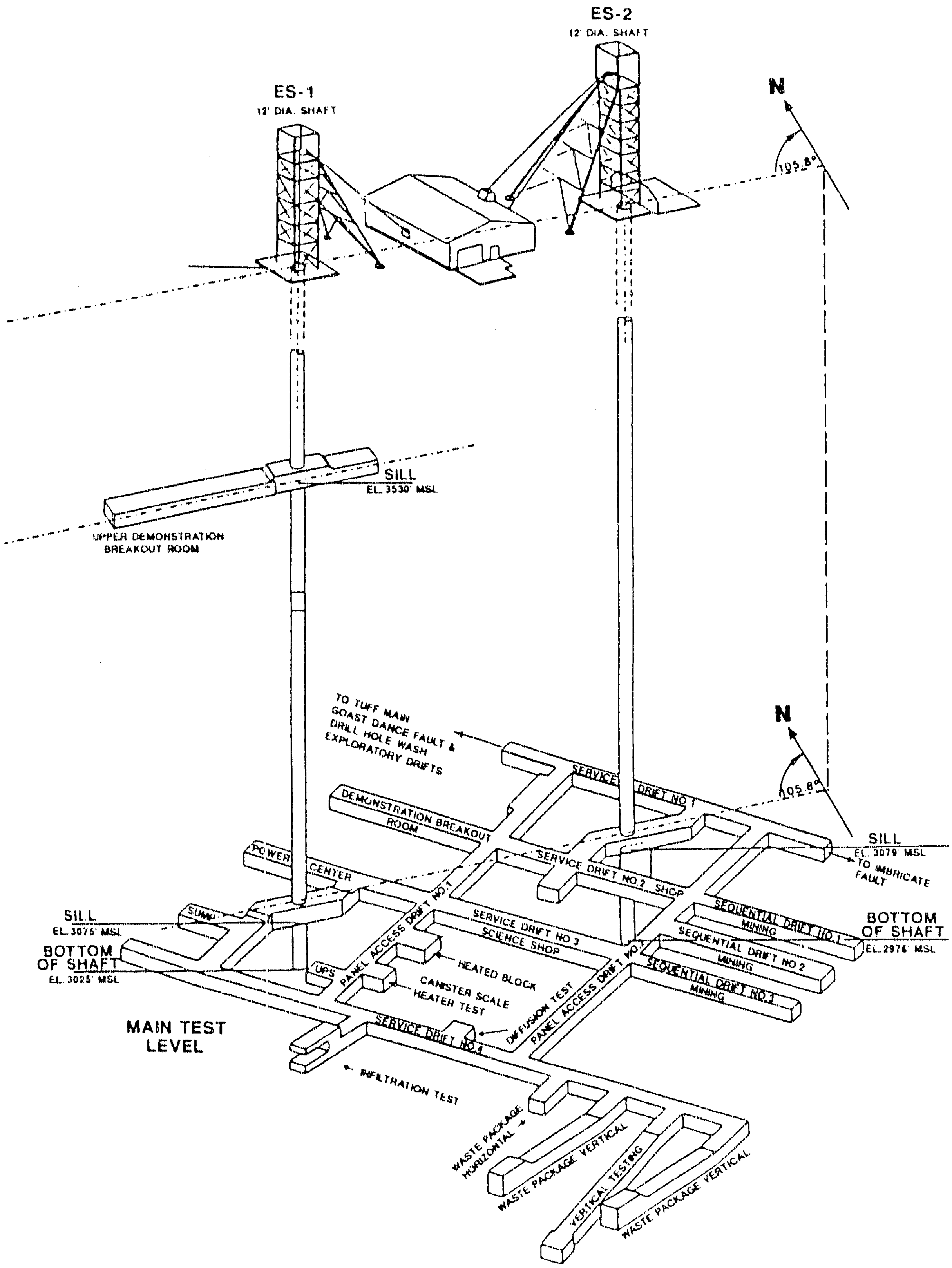


Figure 2-2

Cross Section View of the ESF
(Source: DOE, 1988)
(Drawing No. FS-GA-006, Rev. C)

nonwelded tuff in the Calico Hills Formation. However, this alternative is not currently planned and is not included in the current reference ESF configuration.

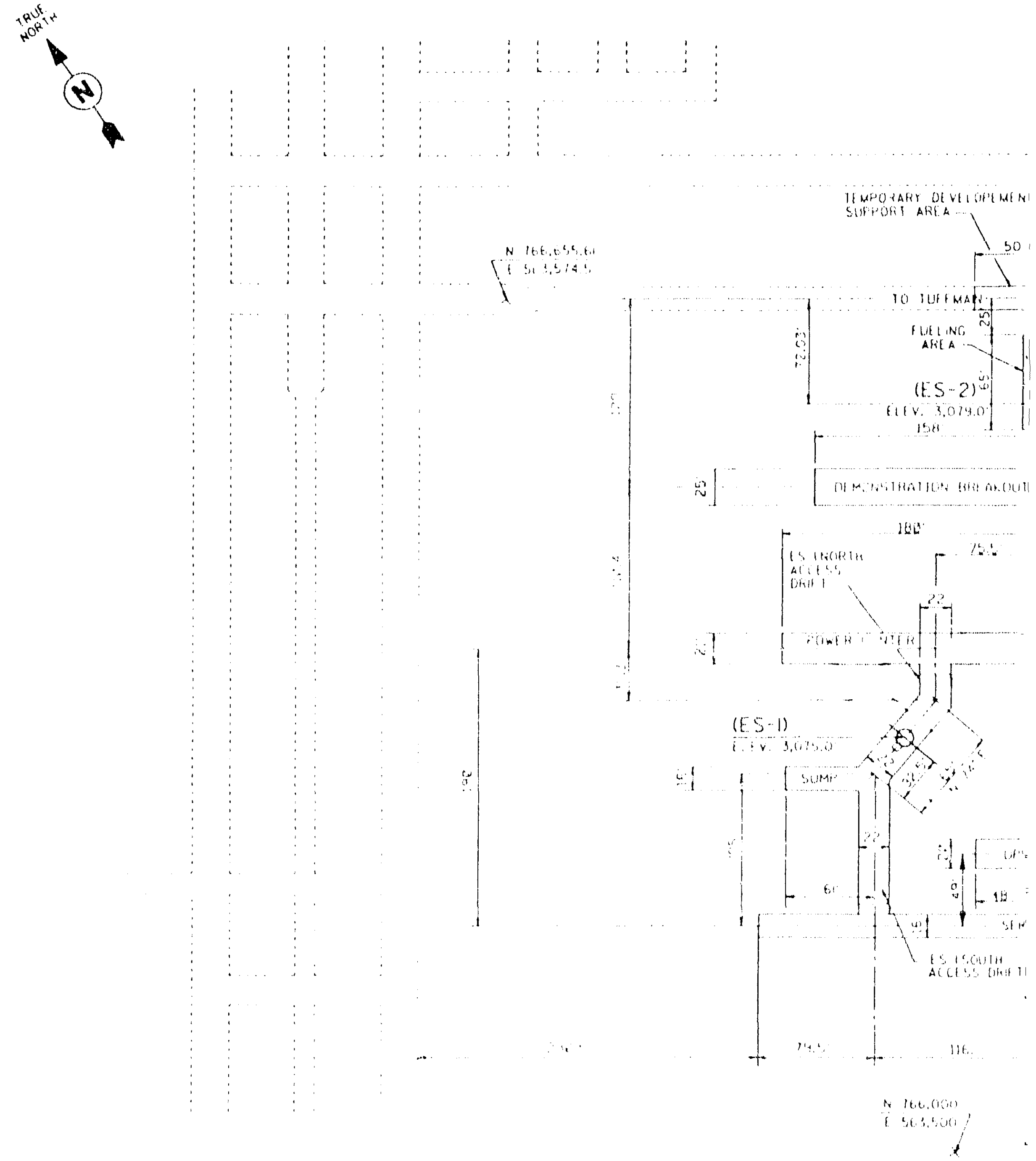
An upper demonstration breakout room, which will be used for testing, will be mined off ES-1 at about 600 ft below the surface.

Approximately 1,050 ft below the surface (at the intended waste emplacement horizon), a main test level will be excavated, including various drifts and alcoves for testing. Figure 2-3 shows the layout of the main test level. Three long exploratory drifts also will be mined at this level to afford access to the Ghost Dance fault, to the Drill Hole Wash structure, and to the imbricate normal fault zone. The drift to the Ghost Dance fault will be about 1,200 ft long to the northwest of the central underground area. The drift to the Drill Hole Wash structure will extend about 1,700 ft to the northeast, and the drift to the imbricate normal fault zone will be about 1,400 ft long to the southeast.

The second exploratory shaft (ES-2) will provide access to the main test level only. It will be used primarily to support construction of the main test level and exploratory drifts.

2.2 Description of Repository During Preclosure Operations

If the Yucca Mountain site is determined to be suitable for a geologic repository, the ESF will be converted to become part of the repository prior to waste emplacement operations. The underground repository facilities will include the ESF excavations as well as additional drifts for access and emplacement of waste.



TRUE NORTH
N
TEMPORARY DEVELOPEMENT
SUPPORT AREA
50
FUELING
AREA
(ES-2)
ELEV. 3,079.0'
158
25
DEMONSTRATION BREAKOUT
180'
ES (NORTH)
ACCESS
DRIFT
22
POWER CENTER
(ES-1)
ELEV. 3,075.0
SUMP
22
ES (SOUTH)
ACCESS DRIFT
79.5
116
N 766,050
E 563,500

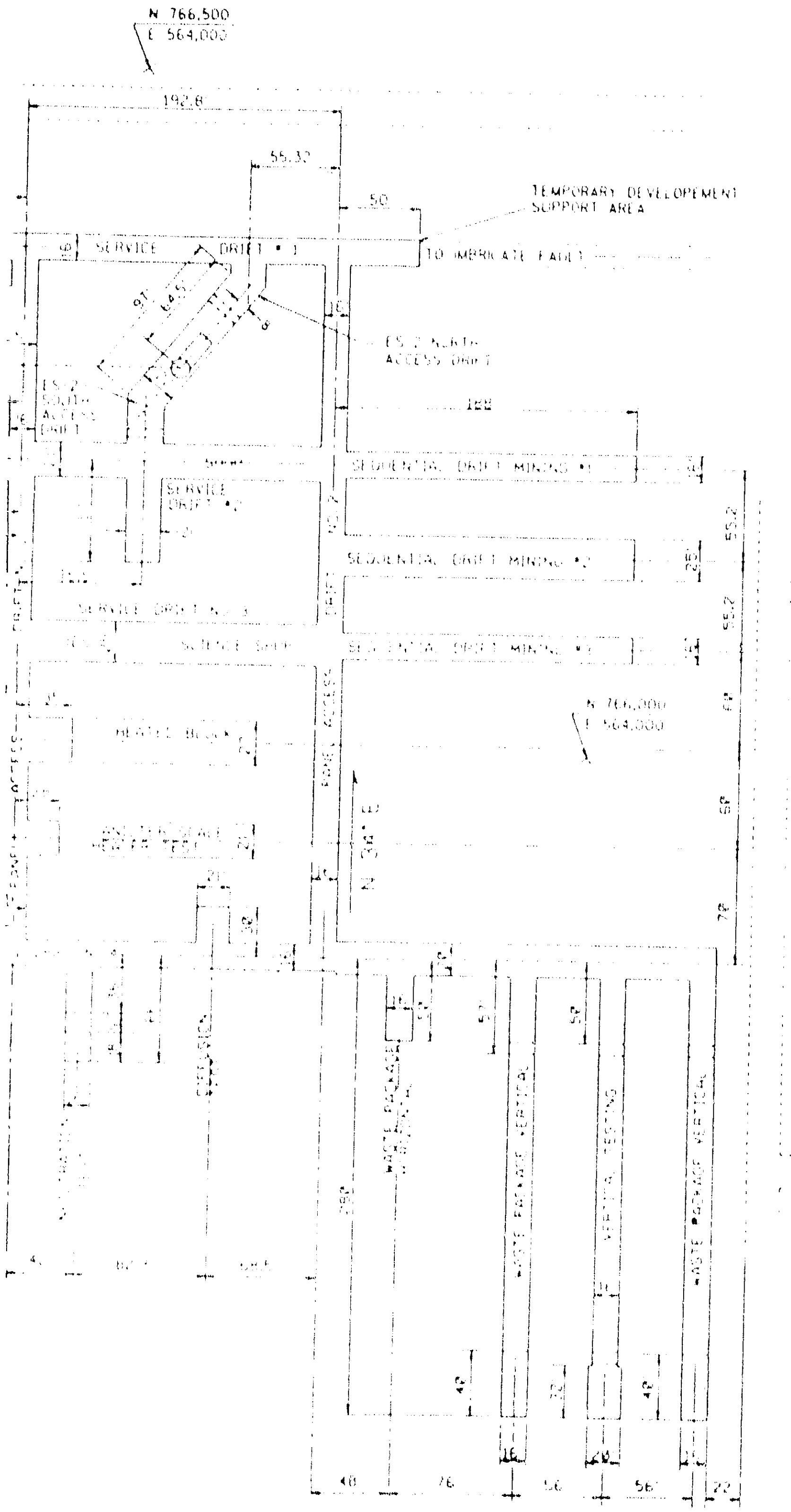


Figure 2-3

Layout of the Main Test Level of the ESF
(Source: DOE, 1988)

(Drawing No. FS-GA 0160, Rev. C)

2.2.1 Converted ESF Configuration

Prior to waste emplacement, the temporary surface facilities of the ESF will be dismantled and removed. The only permanent parts of the ESF to remain during repository operations will be as follows (DOE, 1987):

- Underground openings - the space created by mining or drilling, including those zones within the rock altered by that process

- Shaft liners - all components placed between the inside limits of the shaft and the accessible extent of the underground opening (includes unreinforced concrete, with some steel rods that held the forms used to construct the liners)

- Ground support - any means used to reinforce rock and/or control the movement of rock (e.g., rock bolts), except for removable or replaceable hardware

If operational seals are used in the ESF, they may also remain as part of the repository. However, because the excavations are above the water table, it is not currently expected that these seals will be needed, and they are not included in the reference ESF configuration.

All equipment will be removed from ES-1 and ES-2, leaving completely bare, concrete-lined shafts for ventilation. Surface ventilation structures will be constructed for the repository at the shaft openings (MacDougall, 1987), but no fans will be installed.

Neither ES-1 nor ES-2 will be used to handle radioactive waste at any time (PBQ&D, 1987). The converted ESF main test level will contain no radioactive waste, and although the area will not be needed for repository purposes, it may remain accessible until closure (MacDougall, 1987). Some performance confirmation testing (without radioactive waste) may continue in the main test level during waste emplacement operations.

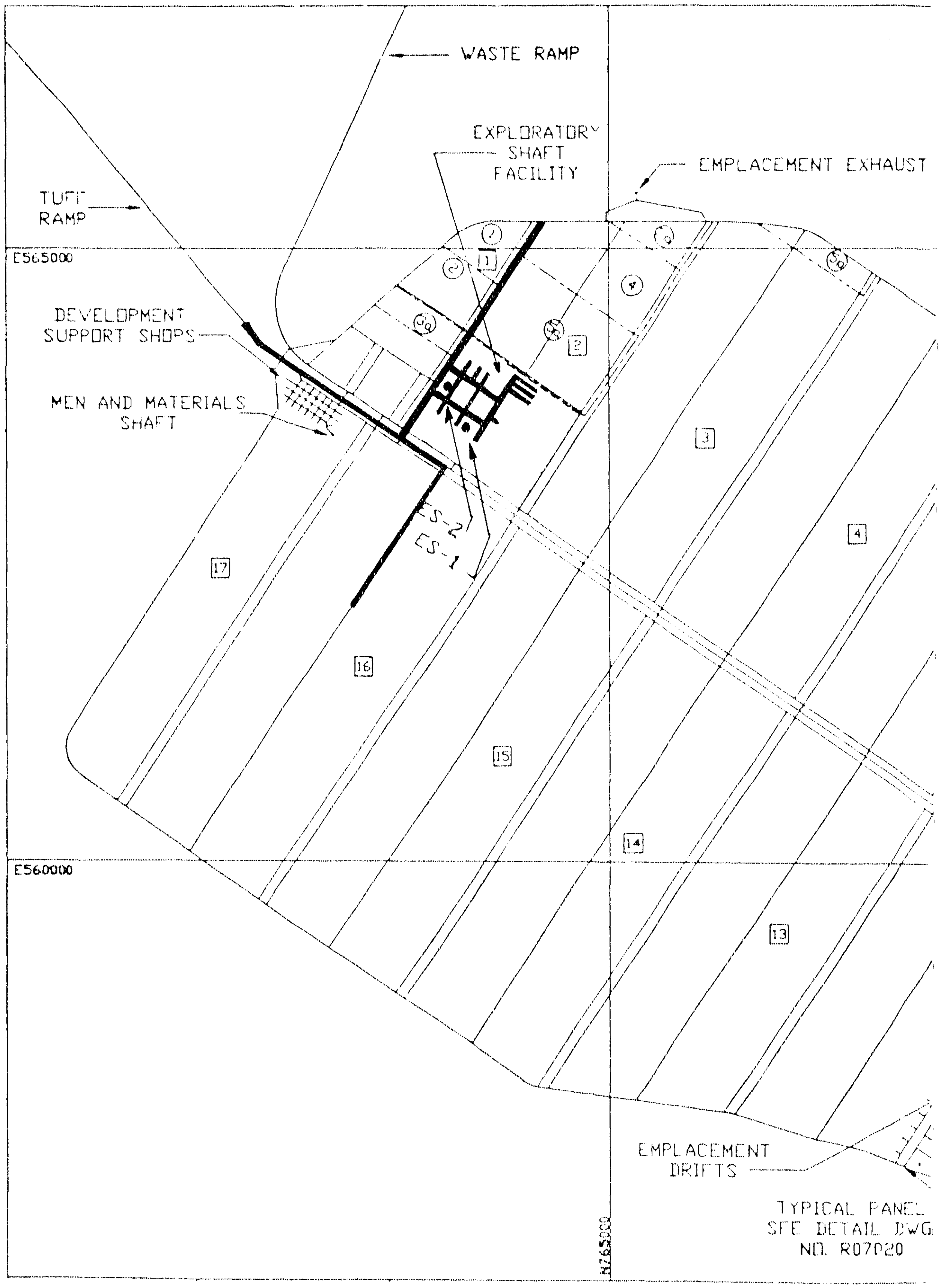
2.2.2 Repository Configuration

The underground repository will surround the main test level of the converted ESF (see Figure 2-4). Two additional shafts will be constructed near the converted ESF, and two ramps will be excavated from the surface to the underground repository.

The two converted exploratory shafts will be used as air intakes for the waste emplacement area during the emplacement phase (MacDougall, 1987). Approximately 60 percent of the total airflow into the waste emplacement area will pass through the two converted exploratory shafts. A waste ramp will permit the waste transporter to move between the surface and underground facilities and will also serve as an air intake for the waste emplacement area. The air will be exhausted from the waste emplacement area through a new emplacement area exhaust shaft. A men-and-materials shaft will provide access for men and materials and will serve as an air intake for the mining development area. A tuff ramp will be used for removing excavated tuff and for routing ventilation exhaust from the mining development area.

An emplacement area exhaust building will be located at the top of the emplacement area exhaust shaft. The building will house motors, exhaust fans, and high-efficiency particulate air (HEPA) filters. The filters will normally be bypassed, unless airborne radioactivity is accidentally released and detected underground.

The three long exploratory drifts extending from the ESF main test level will become part of the network of access drifts for underground areas (see Figure 2-5). The exploratory drift to the Drill Hole Wash structure (to the northeast of the ESF main test area) will become part of the tuff main and ramp, which are


WASTE RAMP
EXPLORATORY SHAFT FACILITY
EMPLACEMENT EXHAUST
TUFF RAMP
E565000
DEVELOPMENT SUPPORT SHOPS
MEN AND MATERIALS SHAFT
ES-2
ES-1
1
2
3
4
13
14
15
16
17
E560000
EMPLACEMENT DRIFTS
TYPICAL PANEL
SEE DETAIL DWG
NO. R07020
N765000

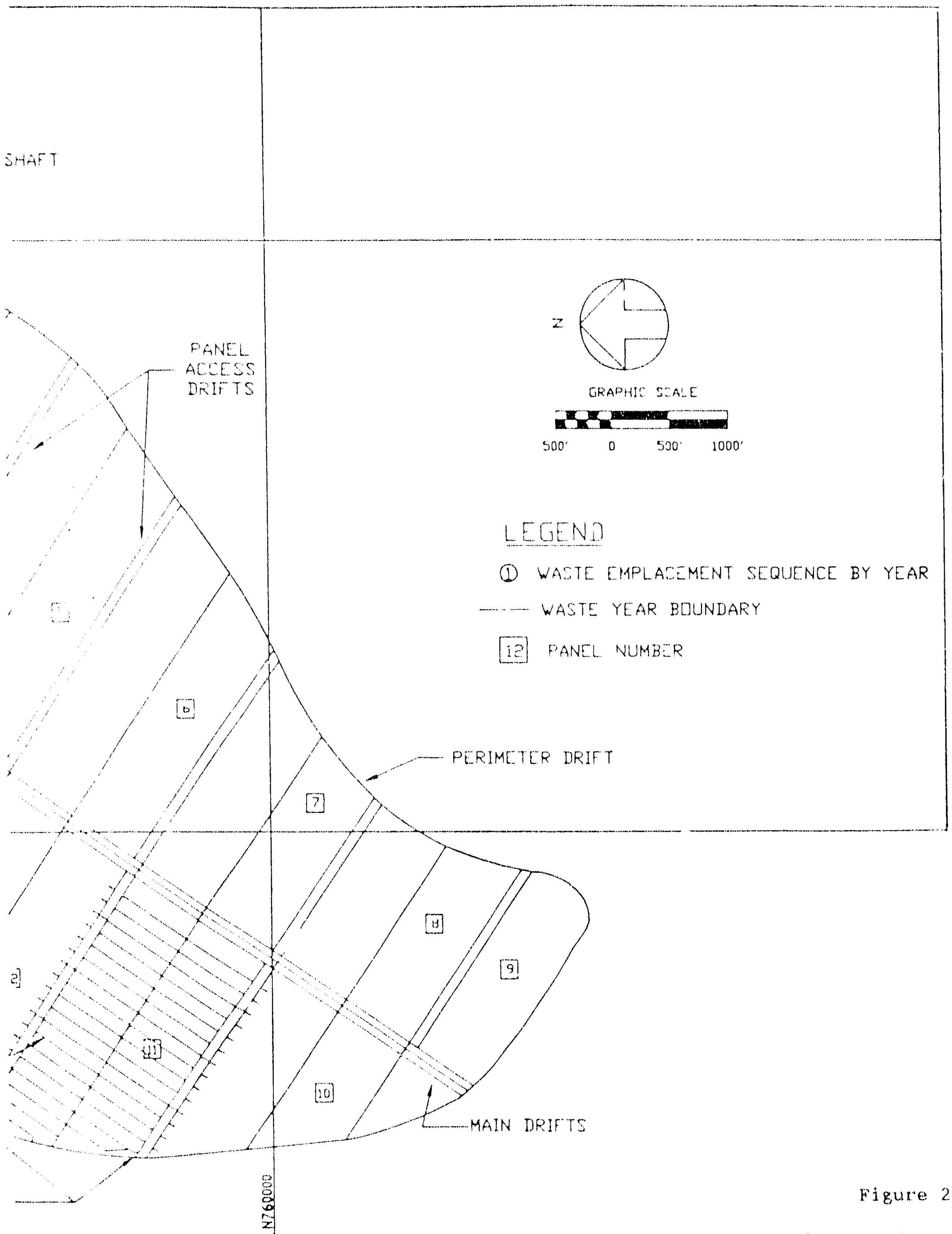
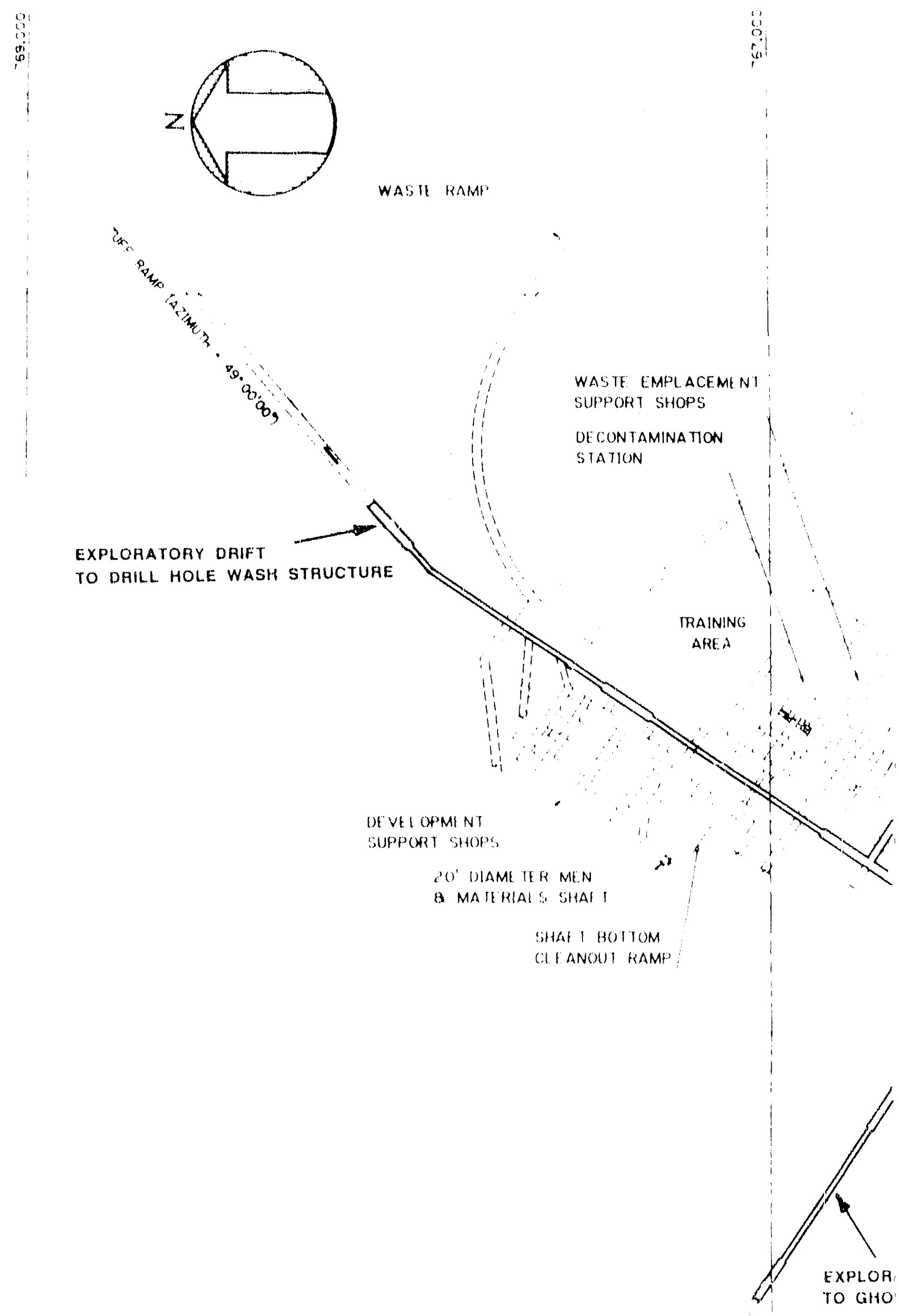


Figure 2-4

Layout of the Underground Repository
(Source: MacDougall, 1987)

(PBQ&D Drawing No. SS/MN632)



KEY:

SOLID LINES DENOTE ESF DEVELOPMENT.

BROKEN LINES DENOTE FUTURE REPOSITORY DEVELOPMENT.

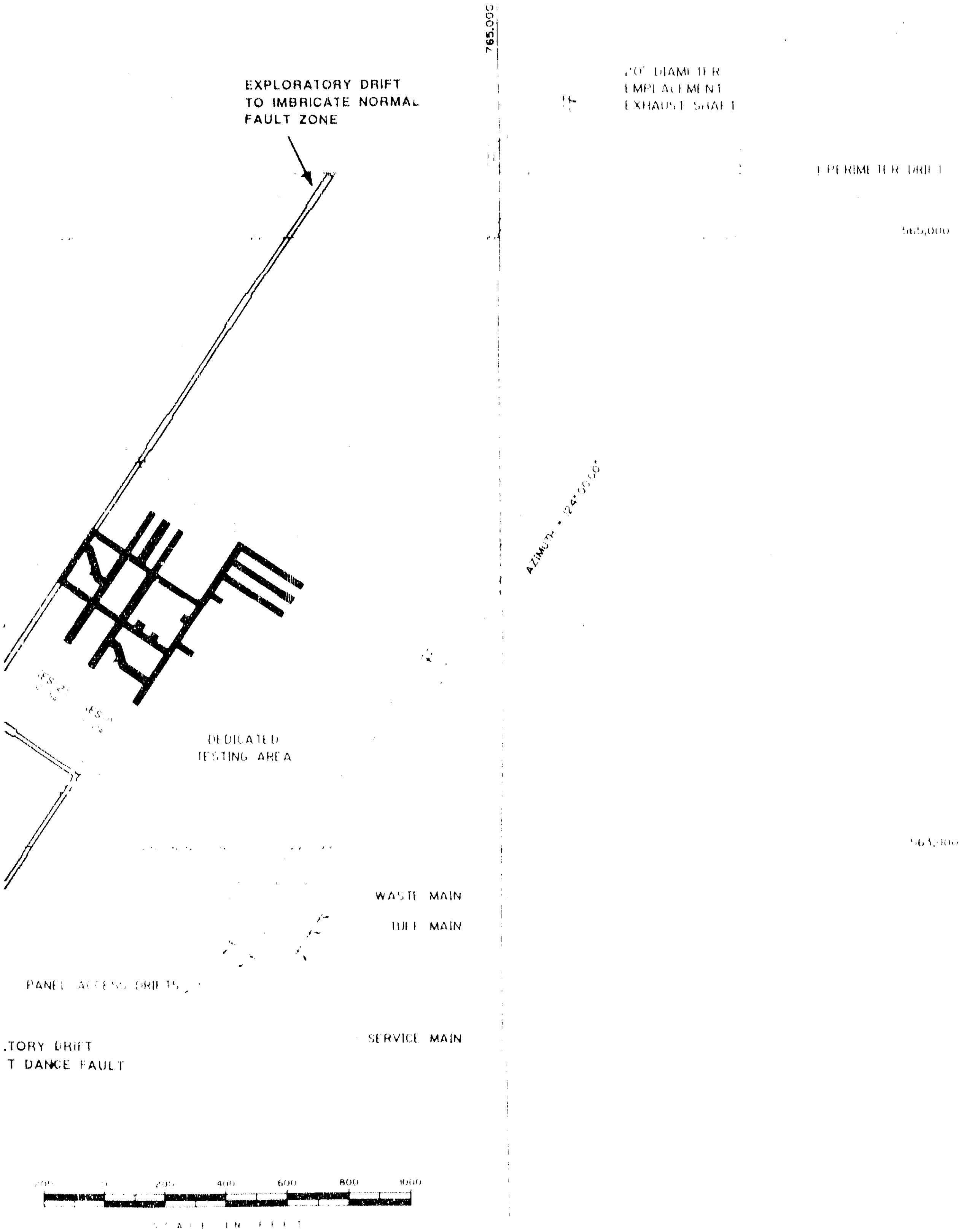


Figure 2-5

Interface Between the ESF and the Underground Repository
During Waste Emplacement
(Source: PBQ&D Drawing SS/MN654A/1, Revision 2)

separated from the waste emplacement area by ventilation barriers. Upon completion of all mining and drilling operations, the associated portion of the tuff main may be used as an alternative pathway for loaded and empty waste transporters. The exploratory drift to the Ghost Dance fault (to the northwest of the main test area) will become a mid-panel drift, which will be used for ventilation and for emplacement of waste at the intersections with emplacement drifts (see Figure 2-6). The exploratory drift to the imbricate normal fault zone (to the southeast of the main test area) will be used for access to waste emplacement areas, but no waste will be emplaced in the drift. The waste transporter will move waste containers through this drift during waste emplacement operations.

During repository operations, the underground facilities will be separated into two areas - the development area and the waste emplacement area. These areas will be separated by ventilation barriers and will have independent ventilation intakes and exhausts. The development area will be maintained at higher pressures than surrounding areas by supply fans, and the emplacement area will be maintained at lower pressures than surrounding areas by exhaust fans.

Additional information on the conceptual design of the underground repository is described by MacDougall (1987).

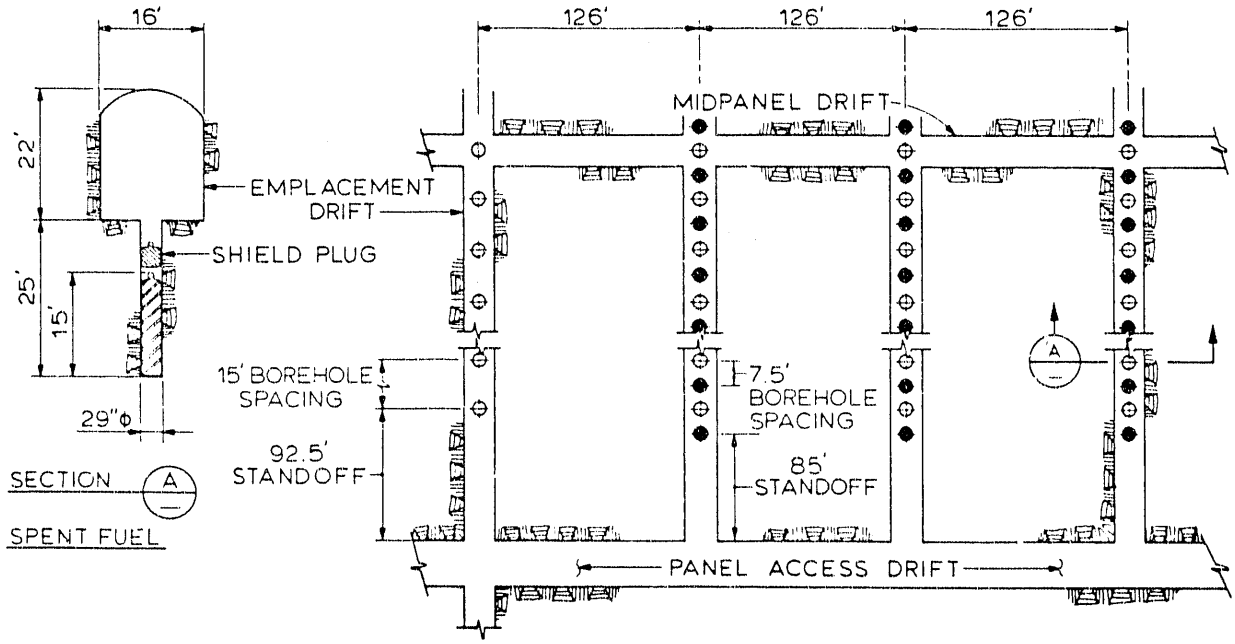


Figure 2-6

Waste Emplacement Configuration
in a Mid-Panel Drift (Exploratory Drift to Ghost Dance Fault)
(Source: MacDougall, 1987)

(PBQ&D Drawing No. SS/MN524)

3.0 BASES FOR THE ASSESSMENT

This section defines "important to safety," summarizes the method used to identify items important to safety, and presents the major assumptions used in this study.

3.1 Definition of "Important to Safety"

10 CFR 60 defines "important to safety," with reference to structures, systems, and components, as "those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure" (NRC, 1986).

The regulations do not specify any numerical frequency or probability of occurrence for such accidents. Procedure AP-6.10Q (Appendix A) state. that only credible accident scenarios having a 0.5 rem or greater offsite dose must be analyzed to identify items important to safety. Thus, for an item to be important to safety, its failure must result in a radiation dose at the site boundary equal to or greater than 0.5 rem and the associated accident scenario must be termed "credible" or be estimated to have a probability greater than 10^{-6}/yr (see Appendix A).

3.2 Method

This assessment follows the method required by DOE Procedure AP-6.10Q, "Identification of Items Important to Safety" (see Appendix A). The method involves 13 steps as described in Table 3-1. The table identifies the subsections of this report in which each step of the assessment for the ESF is discussed.

Table 3-1

PROCEDURAL STEPS FOR
IDENTIFYING ITEMS IMPORTANT TO SAFETY

Step	Description	Subsection
1	Select Documented Design Configuration	2.1 and 2.2
2	Define Compartments	4.1
3	Assign Compartment Locations to Items	4.1
4	Identify and Screen Initiating Events	4.2
5	Develop Event Trees	4.3
6	Estimate Doses	5.1
7	Classify Scenarios as Credible or Not Credible	5.2
8	Identify Credible Scenarios Exceeding the Dose Criterion (Q-Scenarios)	6.1
9	Identify any Other Scenarios Exceeding Other Criteria (Q-Scenarios)	6.1
10	Eliminate NQ-Scenarios from Further Analysis	6.2
11	Evaluate Q-Scenarios to Identify Items Important to Safety	6.3
12	Construct a List of Items Important to Safety	6.3
13	Iterate the Above Steps for Future Stages of the Design	6.4

Source: Appendix A

Step 1 involves the selection of a documented design configuration for the ESF and the repository. These configurations are described in Subsections 2.1 and 2.2.

In Steps 2 and 3, the converted ESF design configuration is separated into compartments. Compartment locations are assigned to each item in the converted ESF (as identified by following the DOE Procedure AP-6.9Q, 1989). These steps are described in Subsection 4.1.

Step 4 involves identification and screening of initiating events. External and internal initiating events are discussed in Subsections 4.2.1 and 4.2.2, respectively.

In Step 5, event trees are developed for credible initiating events that could lead to a significant radiological release. Subsection 4.3 describes the event trees for accidents involving converted ESF failures.

Radiation dose consequences are calculated in Step 6 for each scenario in the event trees. Subsection 5.1 describes the radiation dose calculations.

In Step 7, scenarios are classified as credible or not credible, as described in Subsection 5.2.

Steps 8 and 9 involve identification of credible scenarios that exceed the dose criterion of 0.5 rem or other specified criteria. These scenarios are designated as Q-scenarios. These steps are described in Subsection 6.1.

In Step 10, other accident scenarios that either are not credible or do not exceed the dose criterion are designated as NQ-scenarios and are eliminated from further analyses used to identify items important to safety. This step is discussed in Subsection 6.2.

In Step 11, Q-scenarios are evaluated to identify specific ESF items important to safety. This evaluation is presented in Subsection 6.3.

The list of ESF items important to safety is constructed in Step 12, which is discussed in Subsection 6.3. A list of ESF items that are not important to safety is also included.

Step 13 (the last step) involves iteration of the above steps for future stages of the design. This step and other areas requiring further analyses are described in Subsection 6.4.

This report (and the list of references in Section 7.0) comprises the documentation that demonstrates that each of the above steps has been completed.

3.3 Assumptions

The major assumptions used in this assessment to calculate the doses closely follow those used in previous preclosure radiological safety analyses in Ma (1988) and Appendix F of MacDougall (1987). These major assumptions are listed below; additional assumptions are given in the text.

- The ESF Title I design (DOE, 1988) is used as the reference ESF configuration.

- The SCP-CDR (MacDougall, 1987), which includes the converted ESF, is used as the reference repository configuration for preclosure operations.

- A waste container holds consolidated fuel rods from six PWR fuel assemblies, each with a burnup of 33 GWD/MTU and a cooling period of 10 yr.

- The same fraction of potentially airborne particles is generated for all scenarios involving severe impact on the fuel (such as container drop, runaway transporter, etc.). This fraction (from Appendix F of MacDougall, 1987) is also equivalent to the fraction resulting from a container drop from a height of 35 ft.

- Depletion of airborne radionuclides in the underground facility due to gravitational settling and other deposition mechanisms is neglected.

- The waste transporter cask is designed and constructed using standard industry practices to have sufficient integrity to withstand the effects of drift collapses without releasing radioactivity. This assumption is deemed reasonable because the waste transporter houses the waste container in a thick, strong, shielding cask (about 10 in. steel wall).

4.0 EVALUATION OF POTENTIAL ACCIDENT SCENARIOS

In this section, potential accident scenarios in the converted ESF are developed. To facilitate the assessment, the converted ESF is divided into various compartments (see Step 2 of Table 3-1). Compartment locations are assigned to each ESF item identified as a result of following DOE Procedure AP-6.9Q (see Step 3 of Table 3-1). External and internal initiating events are identified and screened for each compartment (see Step 4 of Table 3-1). Various intermediate events that may occur following each initiating event are then developed and summarized by event trees (see Step 5 of Table 3-1). The assessment closely follows the preclosure radiological safety analyses of the Yucca Mountain repository (Ma, 1988; MacDougall, 1987). Additional details can be found in those reports.

4.1 Compartments of the Converted ESF

To facilitate the assessment, the converted ESF (described in Subsection 2.2) is divided into six compartments (Step 2 of Table 3-1). Each compartment is characterized by the permanently fixed items in it and the operations and functions associated with it during preclosure operations. These compartments are as follows:

1. Main test level
2. Exploratory shafts
3. Drift to the imbricate normal fault zone
4. Drift to the Ghost Dance fault
5. Drift to the Drill Hole Wash structure
6. Upper demonstration breakout room

Another optional compartment is the Calico Hills drill room; however, this is not currently included in the reference ESF configuration and is not considered further in this report.

The compartments are shown in Figure 4-1 and are described in Table 4-1. The table lists the radioactive waste forms that are handled or stored in each compartment and the major operations and functions of each compartment during preclosure operations. Permanent items of the ESF that will remain during preclosure operations are also listed for each compartment (see Subsection 2.2).

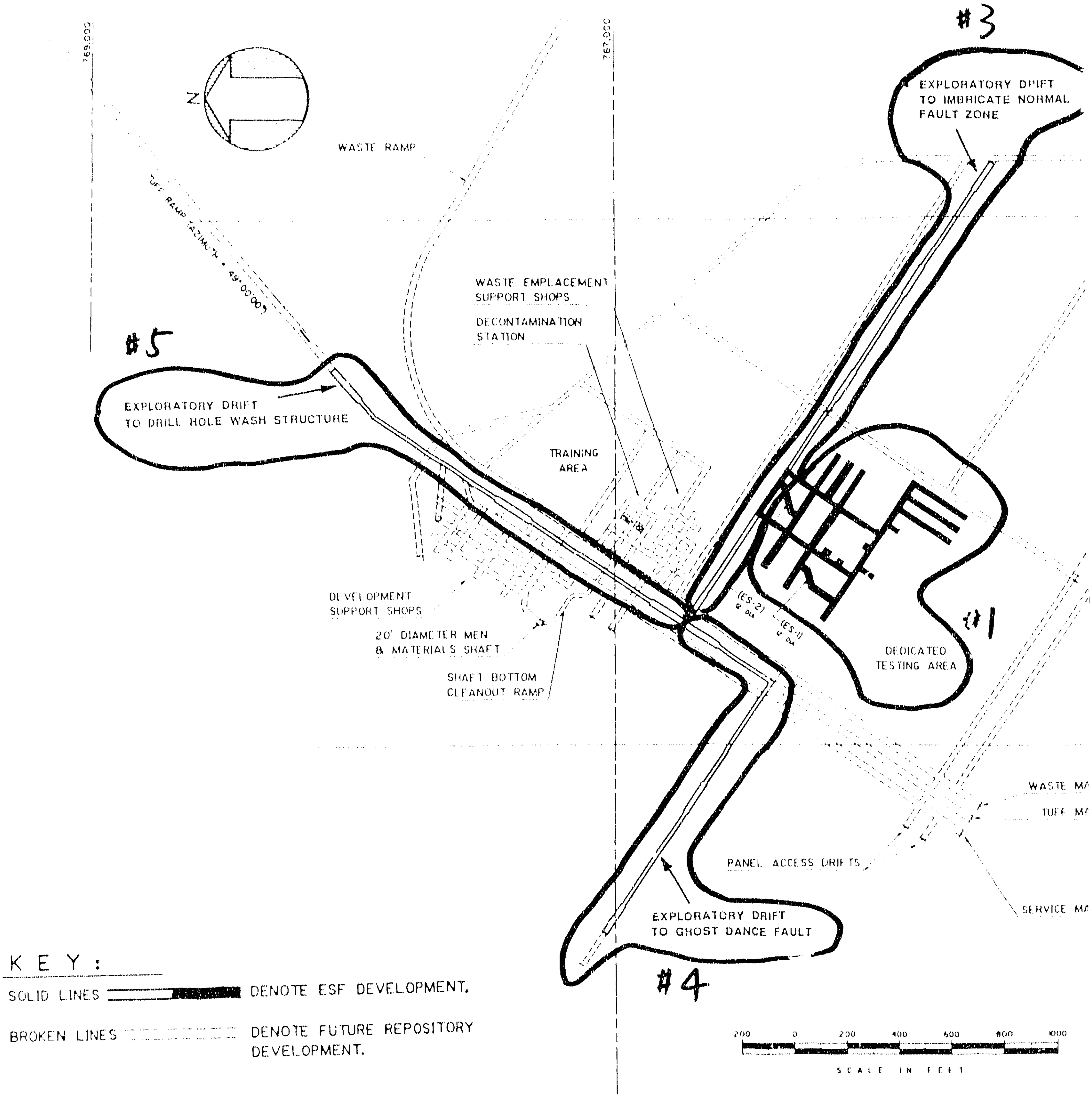
A listing of ESF items from DOE procedure AP-6.9Q (DOE, 1989) and their respective compartment locations is given in Table 4-2.

4.2 Identification and Screening of Initiating Events

In this subsection, both external and internal initiating events are evaluated (see Step 4 of Table 3-1). External events are those caused by natural phenomena or human activities external to the repository. Internal events are those caused by failures (e.g., equipment or structures) or operator activities at the repository.

4.2.1 External Initiating Events

External initiating events that may cause significant radiological releases from the underground facility to the environment are identified and screened in this subsection. The methodology follows that of other preliminary radiological safety analysis (PRSA) reports (MacDougall, 1987; Ma, 1988). A comprehensive checklist for external initiating events, including both natural phenomena and human activities, was compiled based on surveys of previous safety analyses for nuclear facilities (NRC, 1983). This checklist, shown in Table 4-3, is reviewed to determine those events applicable to the site-specific and design-specific conditions. As a result, the number of external events requiring further assessment is significantly reduced.



769,000
767,000
N
WASTE RAMP
TUFF RAMP (AZIMUTH = 49°00'00")
#3
EXPLORATORY DRIFT
TO IMBRICATE NORMAL
FAULT ZONE
WASTE EMPLACEMENT
SUPPORT SHOPS
DECONTAMINATION
STATION
#5
EXPLORATORY DRIFT
TO DRILL HOLE WASH STRUCTURE
TRAINING
AREA
DEVELOPMENT
SUPPORT SHOPS
20' DIAMETER MEN
& MATERIALS SHAFT
SHAFT BOTTOM
CLEANOUT RAMP
(ES-2)
(ES-1)
#1
DEDICATED
TESTING AREA
WASTE MA
TUFF MA
PANEL ACCESS DRIFTS
SERVICE MA
EXPLORATORY DRIFT
TO GHOST DANCE FAULT
#4
KEY:
SOLID LINES DENOTE ESF DEVELOPMENT.
BROKEN LINES DENOTE FUTURE REPOSITORY DEVELOPMENT.
200 0 200 400 600 800 1000
SCALE IN FEET

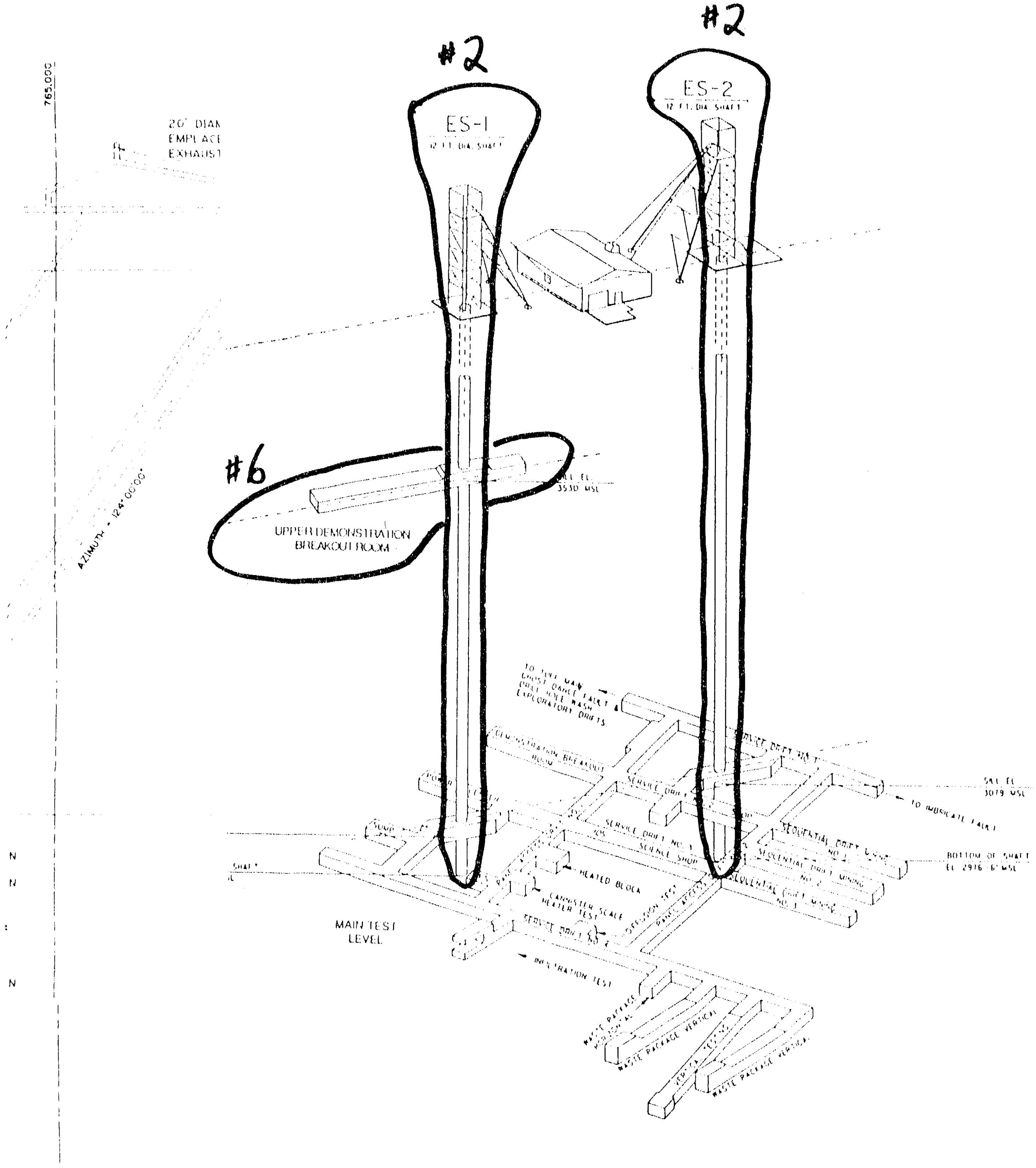


Figure 4-1
Compartments of the ESF

Table 4-1

COMPARTMENTS OF THE ESF

	Compartment	Waste Form During Preclosure Operations	Permanent Items Remaining During Preclosure Operations	Operations and Functions During Preclosure Operations
1.	Main test level	None	Excavations Ground support Operational seals (if needed)	Provide intake airway for waste emplacement area Perform performance confirmation testing
2.	Exploratory shafts (ES-1 and ES-2)	None	Excavations Ground support Shaft liners Operational seals (if needed)	Ventilation intake for waste emplacement area
3.	Drift to imbricate normal fault zone (this drift is a panel access drift during preclosure operations)	Spent fuel in container in transporter cask DHLW in container in transporter cask	Excavations Ground support Operational seals (if needed)	Provide access to waste emplacement drifts Moving loaded transporter and empty transporter Ventilation airway
4.	Drift to Ghost Dance fault (this drift is a mid-panel drift during preclosure operation)	Spent fuel in container in transporter cask DHLW in container in transporter cask Waste container in borehole	Excavations Ground support Operational seals (if needed)	Exhaust airway for emplacement panel Emplacement of waste containers
5.	Drift to Drill Hole Wash fault (this drift is part of the tuff main during preclosure operations)	Spent fuel in container in transporter cask DHLW in container in transporter cask	Excavations Ground support Operational seals (if needed)	Excavated tuff removal Alternative access for equipment and supplies Ventilation exhaust for development area
6.	Upper demonstration breakout room	None	Excavations Ground support Operational seals (if needed)	None

Table 4-2

ESF ITEMS AND THEIR COMPARTMENT LOCATIONS

ESF Item	Item Number	Compartment Number(s) (see Note b)
ESF Site	1.2.6.1	Note a
Main Pad	1.2.6.1.1	Note a
Auxiliary Pads	1.2.6.1.2	Note a
Access Roads	1.2.6.1.3	Note a
Site Drainage	1.2.6.1.4	Note a
Surface Utilities	1.2.6.2	Note a
Power Systems	1.2.6.2.1	Note a
Water Systems	1.2.6.2.2	Note a
Sewage Systems	1.2.6.2.3	Note a
Communication Systems	1.2.6.2.4	Note a
Mine Wastewater Systems	1.2.6.2.5	Note a
Compressed Air Systems	1.2.6.2.6	Note a
Surface Facilities	1.2.6.3	Note a
Ventilation System	1.2.6.3.1	Note a
Test Support Facility	1.2.6.3.2	Note a
Sites for Temporary Structures	1.2.6.3.3	Note a
Parking Areas	1.2.6.3.4	Note a
Material Storage Facilities	1.2.6.3.5	Note a
Shop	1.2.6.3.6	Note a
Warehouse	1.2.6.3.7	Note a
Temporary Structures	1.2.6.3.8	Note a
Communications/Data Building	1.2.6.3.9	Note a
First Shaft	1.2.6.4	2
Collar	1.2.6.4.1	2
Lining	1.2.6.4.2	2
Stations	1.2.6.4.3	Note a
Furnishings	1.2.6.4.4	Note a
Hoist System	1.2.6.4.5	Note a
Sump	1.2.6.4.6	Note a
Second Shaft	1.2.6.5	2
Collar	1.2.6.5.1	2
Lining	1.2.6.5.2	2
Stations	1.2.6.5.3	Note a
Furnishings	1.2.6.5.4	Note a
Hoist System	1.2.6.5.5	Note a
Sump	1.2.6.5.6	Note a
Underground Excavations	1.2.6.6	
Operations Support Areas	1.2.6.6.1	1
Test Areas	1.2.6.6.2	3,4,5,6

Table 4-2 (Cont'd)

ESF Item	Item Number	Compartment Number(s) (see Note b)
Underground Support Systems	1.2.6.7	Note a
Power Distribution System	1.2.6.7.1	Note a
Communications System	1.2.6.7.2	Note a
Lighting System	1.2.6.7.3	Note a
Ventilation Distribution System	1.2.6.7.4	Note a
Water Distribution System	1.2.6.7.5	Note a
Mine Wastewater Collection	1.2.6.7.6	Note a
Compressed Air Distribution	1.2.6.7.7	Note a
Fire Protection System	1.2.6.7.8	Note a
Muck Handling System	1.2.6.7.9	Note a
Sanitary Facilities	1.2.6.7.10	Note a
Monitoring and Warning Systems	1.2.6.7.11	Note a
Underground Tests	1.2.6.8	Note a
Integrated Data Acquisition System (IDS)	1.2.6.8.1	Note a

Notes: a. All ESF items will be removed prior to repository operations except underground openings (shafts and excavations), shaft liners, and ground support.

b. ESF compartments are as follows:

1. Main test level
2. Exploratory shafts
3. Drift to imbricate normal fault zone
4. Drift to Ghost Dance fault
5. Drift to Drill Hole Wash fault
6. Upper demonstration breakout room

c. Identification and numbering of ESF items was accomplished by a task group implementing AP-6.9Q (DOE, 1989).

Table 4-3

CHECKLIST OF EXTERNAL INITIATING EVENTS

Seismic activity (faulting, shear zone)
Flooding (storm, river diversion)
Lightning
Volcanic activity
Weather fluctuations and extremes (fog, frost, hail, drought, high temperature, low temperature, rapid thaws, ice cover, snow, etc.)
Chemical effects (release of chemicals or toxic gas)
Sandstorm
Tornado
Extreme wind
Industrial activity induced accident
Military accident (weapons testing, aircraft impacts, bombing range)
Crash of a commercial aircraft (helicopter, passenger planes, etc.)
Undetected past intrusion (undiscovered boreholes or mine shafts)
Inadvertent future intrusion
Fire (forest fire, brush fire)
Pipeline accident (gas, etc.)
Loss of offsite power
Perturbation of groundwater system (establishment of population center, reservoirs, irrigation, intentional artificial recharge, etc.)
Avalanche
Static fracturing (surficial fissuring, impact fracturing, hydraulic fracturing)
Denudation and stream erosion
Magmatic activity (extrusive, intrusive)
Coastal erosion
High tide, high lake level, or high river stage
Low lake or river water level
Hurricane
Meteorite impact
Seiche
Tsunami
Dam failure
Waves
Undetected features and processes (breccia pipes, lava tubes, gas or brine pockets, etc.)
Sedimentation
Subsidence
Landslide
Uplifting
Thermal loading (differential elastic response, nonelastic response, fluid pressure changes, fluid migration, canister migration, etc.)
Geochemical alterations
Waste and rock interactions
Rock deformation
Glaciation
Dissolution
Epeirogenic displacement (igneous emplacement, isostasy)
Orogenic diastrophism (near-field faulting, far-field faulting, diapirism, diagenesis)
Rockburst

Sources: NRC (1983)

External events were eliminated if they met one or more of the following screening criteria:

- The external event is either not applicable to the Yucca Mountain underground facilities or not credible.
- The external event is irrelevant to preclosure operations or insignificant during this period.
- The external event will not cause any significant radiological releases to the environment.
- The impact of the external event is well within the plant design basis and does not need to be considered for this study.

The screening of the external events given in Table 4-3 was performed on a compartment-by-compartment basis. The results are shown in Table 4-4. Some discussions of the screening are given below. Additional discussions can be found in previous PRSA reports (Ma, 1988; MacDougall, 1987).

Flooding is not selected for detailed assessment because the tops of the shafts are well above the probable maximum flood (PMF) level, as shown in Figure 4-2 (DOE, 1988).

An undetected geologic fault is not selected for detailed assessment, since it is not credible that a waste container would be placed in a borehole with a major undetected fault. On the other hand, in a borehole with minor undetected geologic features (such as joints) is considered possible, and as a result, rocks may slide into this borehole. Because the emplacement container is made of 3/8 in. thick steel, movable rocks that slide into the borehole may damage the container, but will not cause it to breach. Therefore, no significant radiological release will result.

Table 4-4

RESULTS OF EXTERNAL INITIATING EVENT SCREENING

Events Selected for Consideration

Seismic activity

(1) Events Not Applicable to Underground Facilities or not Credible

Aircraft crash (commercial and military)
Flooding
Pipeline accident
Avalanche
Coastal erosion
High tide, high lake or river level
Low lake or river level
Hurricane
Meteorite impact
Seiche
Tsunami
Dam failure
Waves
Sedimentation
Dissolution
Static fracturing
Stream erosion
Undetected geologic fault
Undetected features and processes (breccia pipes, lava tubes, gas or brine pockets)
Chemical effects
Industrial activity accident
Volcanic activity

(2) Events Irrelevant to or Insignificant During Preclosure Operations

Inadvertent future intrusion
Orogenic diastrophism
Glaciation

Subsidence
Uplifting
Magmatic activity
Epeirogenic displacement

Events with Insignificant Radiological Consequences

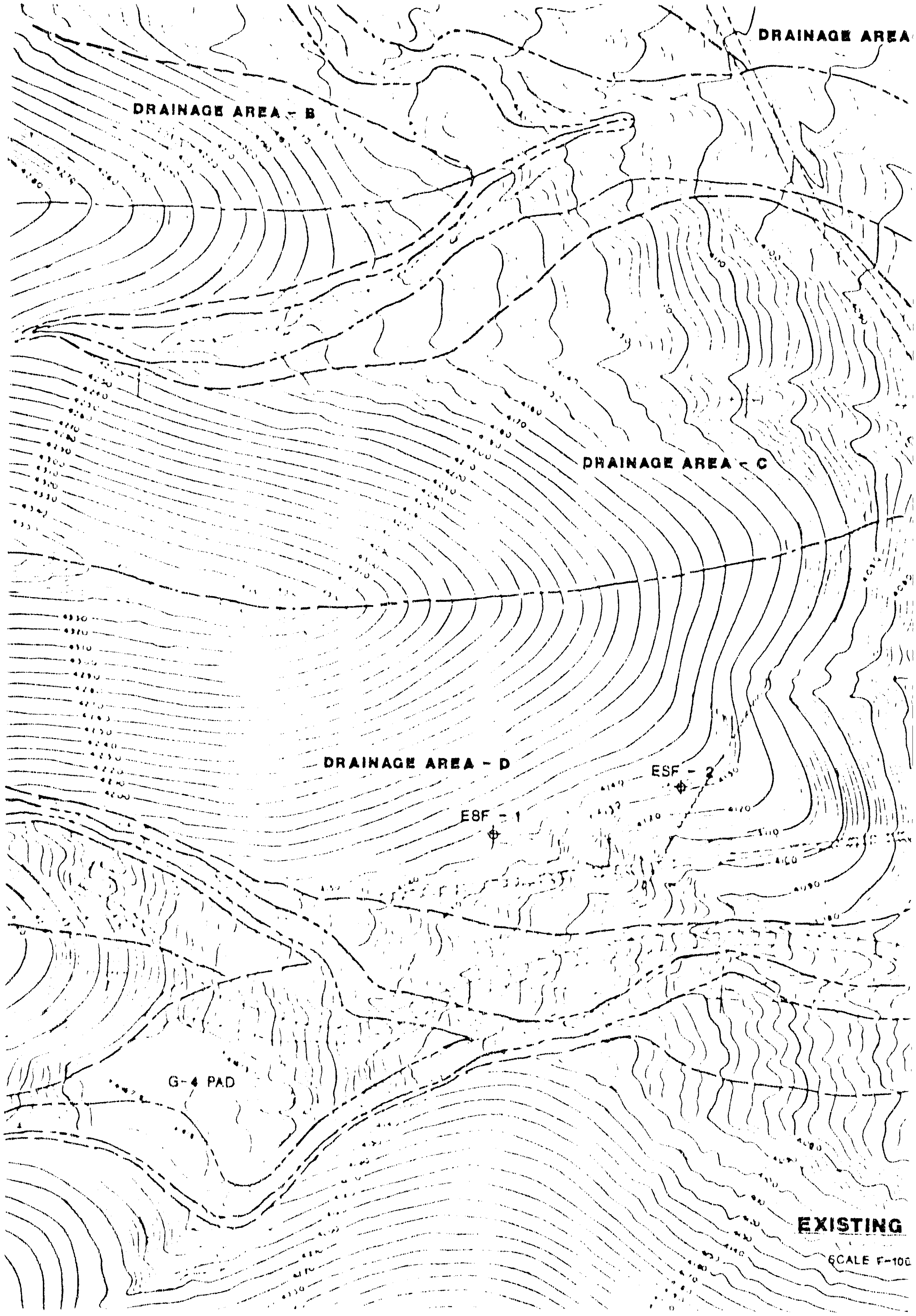
Loss of offsite power
Undetected geologic features (joints)[a]
Weather fluctuations and extremes
Extreme wind
Tornado
Sandstorm
Lightning
Undetected past intrusion
Forest fire or range fire
Landslide
Rockburst
Groundwater perturbation

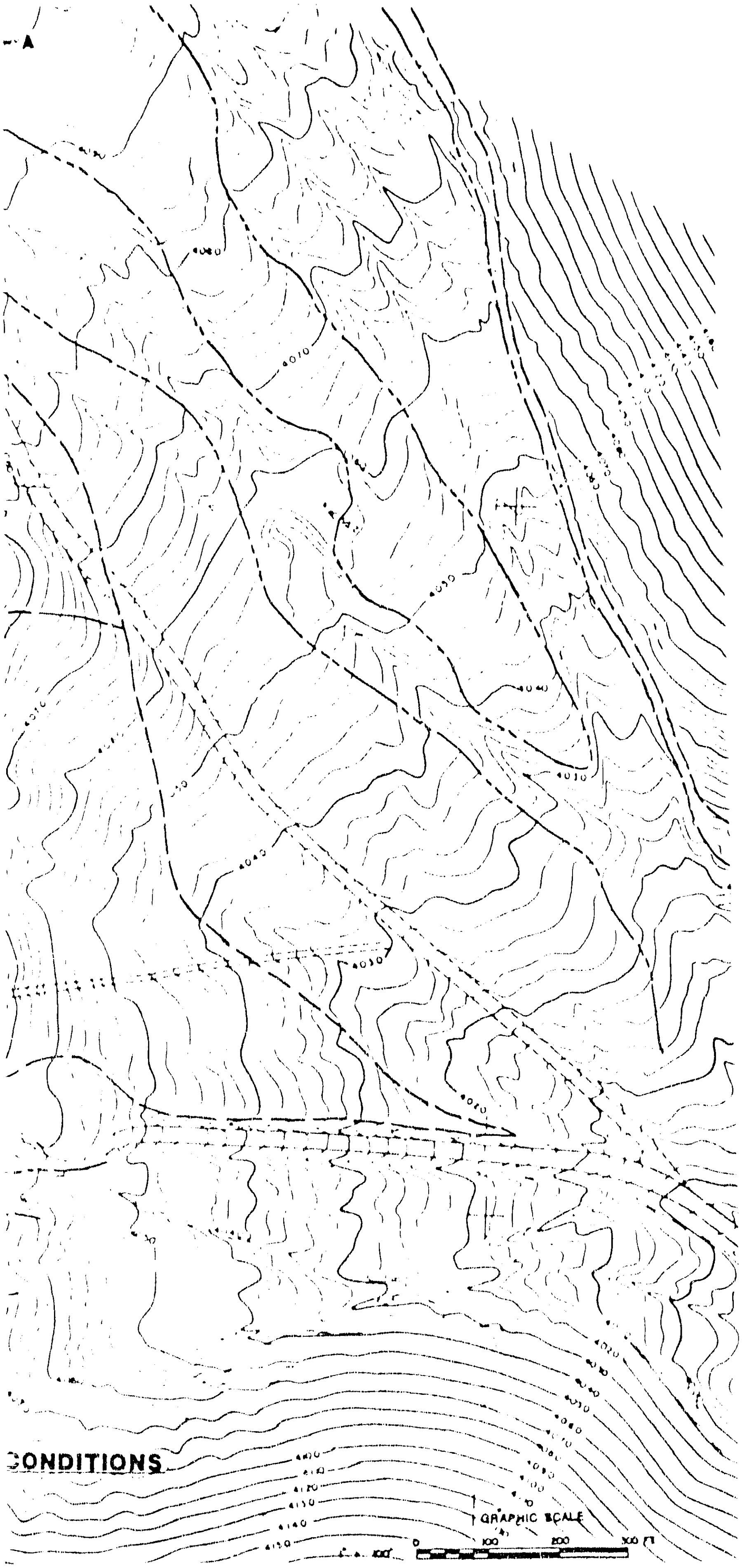
Events with Impact Considered Within the Repository Design Basis

Thermal loading[b]
Waste and rock interactions
Rock deformation
Geochemical alteration

(a) Undetected features (joints) include the formation of movable blocks that slide into boreholes.

(b) The heat generated by the waste forms will not rupture or crack the containers.


DRAINAGE AREA
DRAINAGE AREA - B
DRAINAGE AREA - C
DRAINAGE AREA - D
ESF - 2
ESF - 1
G-4 PAD
EXISTING



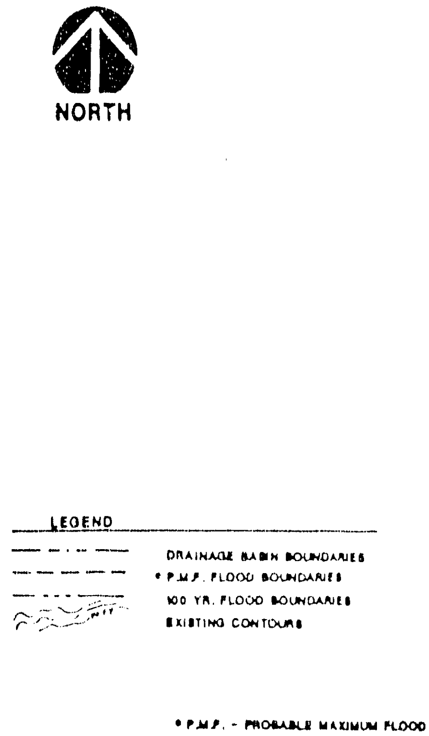


Figure 4-2

Probable Maximum Flood Levels
Near the Exploratory Shafts
(Source: DOE, 1988)

(Drawing No. JS-025-ESF-C45)

For nuclear power plants licensed by NRC, chemical effects do not require evaluation if there are no major storage areas or shipments of hazardous chemicals within 5 mi of the facility (NRC, 1974a). Because the Yucca Mountain repository satisfies this criterion, chemical effects are not selected for further assessment.

Brush or range fires are considered possible. Smoke resulting from these fires may enter the intake airway of the converted exploratory shafts; as a result, the underground ventilation system may be shut down. However, since this event will not lead to any radiological release underground, brush or range fires are not selected for further assessment.

Localized landslide near the tops of the exploratory shafts is considered possible. The debris may plug the shaft and reduce ventilation flow, but no underground release will result. This event is similar to the collapse of shaft concrete liners, which will be analyzed in detail as an internal initiating event. Thus, landslide is not selected for further assessment.

Aircraft crash accidents are not selected for detailed assessment for the converted ESF. There are two major reasons for this: (1) there is no commercial airport nearby and (2) a preliminary study currently under way indicates that the probability of a military aircraft crash onto the repository site is about $10^{-12}/ft^2$-yr. Because the area of the two converted exploratory shafts is about 10^3 ft^2, the annual probability of a military aircraft crash onto the exploratory shafts is about 10^{-9}/yr, which is not credible.

The effects of underground weapons testing at the nearby Nevada Test Site is not selected for detailed assessment, because preliminary studies indicate that the peak horizontal ground accelerations due to underground nuclear explosions are smaller than the design basis seismic ground motion.

Volcanic activity is not expected at the Yucca Mountain site (DOE, 1986), and is not selected for further assessment. This topic will be studied further during site characterization.

The results of the screening in Table 4-4 indicate that only the seismic event is selected for detailed assessment.

4.2.2 Internal Initiating Events

Internal initiating events in the underground facilities that are credible and have the potential of causing significant radiological releases to the environment have been identified and screened in Ma (1988). Two methods were used to identify internal initiating events: (1) survey forms, which document accidents judged to be credible by experienced designers for each compartment, and (2) interaction matrices, which identify possible interactions between items in each compartment. This same approach is adopted in this assessment to identify internal initiating events. Based on previous analyses (Ma, 1988; MacDougall, 1987), the initiating events selected for consideration include the following:

- Structural collapses, such as a collapse of a drift roof along a length that is comparable to the drift width (i.e., about 20 ft), or a shaft liner collapse

- Underground transporter accidents, such as transporter collision, transporter slide, or runaway transporter

- Accidents during emplacement or removal of containers, such as a grapple or transporter hoist failure and container failure

Among these initiating events, structural collapse of a drift or shaft liner is the only failure mode of the converted ESF that requires further assessment. The other two types of events (transporter accidents and container handling accidents) may occur in the converted ESF, but are not related to any permanent items of the ESF. These other two types of events are addressed when considering radiological releases from other underground areas.

Failures of operational seals (if any) could result in some water leakage into the underground repository; however, because the repository is above the water table, the amount of water leakage will be relatively small. Therefore, failures of the seals will not cause any radiological releases and will not affect any other accidental releases from other areas of the underground repository.

In case of a radiological release underground, the release of radioactive airborne particles to the atmosphere will be reduced by the following two design features of the underground ventilation system:

- Upon detection of high airborne radioactivity concentrations, the exhaust air is routed through HEPA filters housed in the surface building of the waste emplacement exhaust shaft.

- Any air leakage between ventilation systems is in the direction of the waste emplacement area; thus, released materials will move toward the emplacement ventilation exhaust system, not the development ventilation exhaust system.

In assessing the converted ESF failures, the major question is whether the failure could result, either directly or indirectly, in a radiological release with an offsite dose equal to or greater than 0.5 rem.

The results of this assessment are described below and summarized in Table 4-5.

Table 4-5

POTENTIAL FAILURES OF THE CONVERTED ESF AND THEIR EFFECTS

	Compartment Area	Failure Mode	Effects
1.	Main test level	Drift collapse	Will not cause any radiological releases May liberate excessive dust that could plug the HEPA filters and/or make radioactivity monitors inoperable in the event of concurrent radiological releases
2.	Exploratory shafts	Concrete liner collapse	Will not cause any radiological releases May generate excessive dust (although unlikely) that could plug the HEPA filters and make radioactivity monitors inoperable in the event of concurrent radiological releases
3.	Drift to imbricate normal fault zone	Drift collapse	Will not cause any radiological releases May liberate excessive dust that could plug the HEPA filters and/or make radioactivity monitors inoperable in the event of concurrent radiological releases
4.	Drift to Ghost Dance fault	Drift collapse	May cause radiological releases Will not affect radiological releases from other areas
5.	Drift to Drill Hole Wash fault	Drift collapse	Will not cause any radiological releases Will not affect radiological releases from other areas
6.	Upper demonstration breakout room	Drift collapse	Will not cause any radiological releases Will not affect radiological releases from other areas

4.3 Development of Event Trees

4.3.1 Intermediate Events

In this subsection, the intermediate events that follow the initiating events identified above are described for each compartment (see Step 5 of Table 3-1). The effects of these intermediate events are also described.

4.3.1.1 Main Test Level. Drift collapse is the only significant structural failure for the main test level. The length of the drift collapse is assumed to be the same as the drift width, which is about 20 ft. Because the main test level does not contain any radioactive waste (see Table 4-1), no radiological release will result.

A drift collapse may completely block the intake airway of the two exploratory shafts. However, in this very unlikely event, the waste ramp will still provide the ventilation intake for the waste emplacement area. Additionally, the waste emplacement area will continue to be maintained at a lower pressure with respect to the mining development area. It should be noted that the ventilation air pressure in the waste emplacement area will be lower with the two shafts closed than with them open. Consequently, air from the waste emplacement area will continue to exhaust through the emplacement exhaust shaft, and any air leakage will continue to be from the mining development area to the waste emplacement area.

If a radiological release occurs in other areas of the underground facility, the exhaust air will be routed through HEPA filters, as long as the airborne radioactivity monitoring system and the airflow control system perform properly. However, if a drift collapse occurs at the same time, dust liberated from the drift surfaces may be carried by the large airflow and (1) eventually plug the HEPA filters or (2) plug filters in the airborne radioactivity monitors, rendering them inoperable. Plugging of the HEPA filters can cause the ventilation system to fail in one of two ways:

- The waste emplacement area may no longer be maintained at a negative pressure, and some air may leak through the waste ramp and exploratory shafts without filtration.

- Significant plugging could damage the HEPA filters (although unlikely), resulting in the unfiltered release of airborne radioactive particles.

Plugging of the airborne radioactivity monitor filters may result in failure to activate the standby HEPA filters in the emplacement exhaust system, which may cause radioactive airborne particles to bypass the HEPA filters and to be released to the atmosphere without filtration.

In either case - plugging the HEPA filters or bypassing the HEPA filters - the ventilation system would not filter radiological releases.

In the present study, the amount of dust generated by a drift collapse is not determined, nor is the transport of airborne particles through long underground tunnels analyzed. The plugging or bypassing of the HEPA filters is, however, considered as a possible scenario for the main test level.

The main test level may be used for performance confirmation testing during the waste emplacement period. Therefore, some combustible materials may be present, and fire can be regarded as a potential accident. In the event of a fire, no radioactive materials will be released since no radioactive waste is contained in this area. Detailed fire scenarios involving electrical fires, mechanical equipment fires, and chemical explosion are not developed because of the current lack of design details. However, it is assumed that only very small amounts of combustible materials will be in this area; thus, if a fire breaks out, it will be very small and will not affect any radiological releases. As design details become available, further analysis of fire hazards will be required.

4.3.1.2 Exploratory Shafts. The most severe structural failure of the exploratory shafts (both ES-1 and ES-2) is the collapse of the concrete liner with a vertical dimension of about 20 ft. The collapse could be spontaneous or could be caused by an earthquake; such a collapse, however, is considered to be very unlikely or not credible (the embedded steel hanging rods also provide some additional partial reinforcement). Because no radioactive waste is handled in the exploratory shafts, a concrete liner collapse will not lead to any radiological releases.

The previous discussion of the airflow path for the drift collapse in the main test level also applies to this case. Even the complete blockage of the two exploratory shafts (an unlikely event) will not disrupt the ventilation system of the waste emplacement area. The waste emplacement area will continue to be maintained at a lower pressure than surrounding areas (e.g., the development area).

If a shaft liner collapses, concrete pieces falling from a height of about 1,000 ft will be fractured, but no substantial amount of dust will be generated. Therefore, if a radiological release occurs from another area at the same time, it is unlikely that the dust will plug the HEPA filters or make airborne radioactivity monitors inoperable, and cause the ventilation filtration system to fail.

Because the repository and shafts are above the water table, and any small amounts of water will seep through the construction joints (and not accumulate behind the shaft liners), additional water inflow into the shaft due to liner collapse is not a concern in this assessment.

4.3.1.3 Drift to Imbricate Normal Fault Zone. Drift collapse is the only significant structural failure in this area. During preclosure operations, this drift serves as an access to waste emplacement drifts for the waste transporter and also as a ventilation airway. The transporter cask has not yet been designed, but it is assumed that the cask thickness will be at least 10 in. of steel for radiation shielding. With this thick and strong shielding, the transporter cask will be able to withstand a drift collapse without major distortion. In addition, spent fuel and DHLW canisters are further protected by a 3/8 in. steel container inside the cask. It is therefore concluded that no radioactivity will be released as a result of drift collapse in this compartment.

Although a drift collapse in this area may partially block the ventilation airway, this partial blockage will have an insignificant effect on the air intake and air exhaust for the waste emplacement area. However, if drift collapse and radiological releases occur at the same time, the dust liberated from the drift surfaces may plug the HEPA filters or make the airborne radioactivity monitors inoperable. As a result, the ventilation filtration system for the emplacement area could fail.

4.3.1.4 Drift to Ghost Dance Fault. Drift collapse is the only significant structural failure in this area. During preclosure operations, this drift will be a mid-panel drift, which serves as an exhaust airway for emplacement drifts in the panel. Loaded and empty waste transporters will cross the mid-panel drift. In addition, waste containers may be emplaced in boreholes at the intersections of the emplacement drifts and the mid-panel drift. Because radioactive waste is handled in this area, a drift collapse onto a loaded transporter during emplacement or removal of containers may cause a container drop or transporter slide, which may result in radiological releases.

A drift collapse in this area may partially block the ventilation airway, but this partial blockage will have an insignificant effect on the air intake and air exhaust for the waste emplacement area.

The drift collapse may liberate dust from the drift surfaces. It is judged, however, that the drift is sufficiently far away from the emplacement exhaust shaft so that the dust will settle to the ground before reaching the exhaust system. As a result, the HEPA filters and the airborne radioactivity monitors will not be significantly affected. Thus, in the event of a concurrent underground radiological release, the ventilation system will continue to function properly.

4.3.1.5 Drift to Drill Hole Wash Fault. Drift collapse is the only significant structural failure in this area. This drift is part of the tuff main and tuff ramp during preclosure operations. Upon completion of all mining and drilling operations, the associated portion of the tuff main may be used as an alternative pathway for loaded and empty waste transporters. Because of the strong construction of the transporter cask, a drift collapse in this area will not result in any radiological releases. In addition, this drift is sufficiently far from the emplacement exhaust shaft so that any dust liberated by the drift collapse will settle to the ground before reaching the exhaust system. Thus, a drift collapse in this area will not affect any other concurrent radiological releases in the waste emplacement area.

4.3.1.6 Upper Demonstration Breakout Room. The upper demonstration breakout room at elevation 3,530 ft is not involved in any preclosure operations. Therefore, a drift collapse in this area will not cause any radiological releases and will not affect any other concurrent radiological releases.

4.3.1.7 Conclusions. Based on the above discussions, the effects of structural failures of the converted ESF can be summarized as follows:

- A drift collapse on a loaded transporter during emplacement or removal of waste containers in the drift to Ghost Dance fault may result in radiological releases.

- A drift collapse in the main test level and in the drift to imbricate normal fault zone may liberate dust from the ground surface and cause the filtration system to fail. If a radiological release occurs at the same time in other areas of the underground facility, the radioactive airborne particles could be released to the atmosphere without filtration.

- In the very unlikely event of a concrete liner collapse in the exploratory shaft, no substantial amount of dust will be generated. Thus, it is unlikely that the filtration system would fail as a result of shaft liner collapse.

Structural failures in other compartments are not analyzed further, since they do not lead to any radiological consequences.

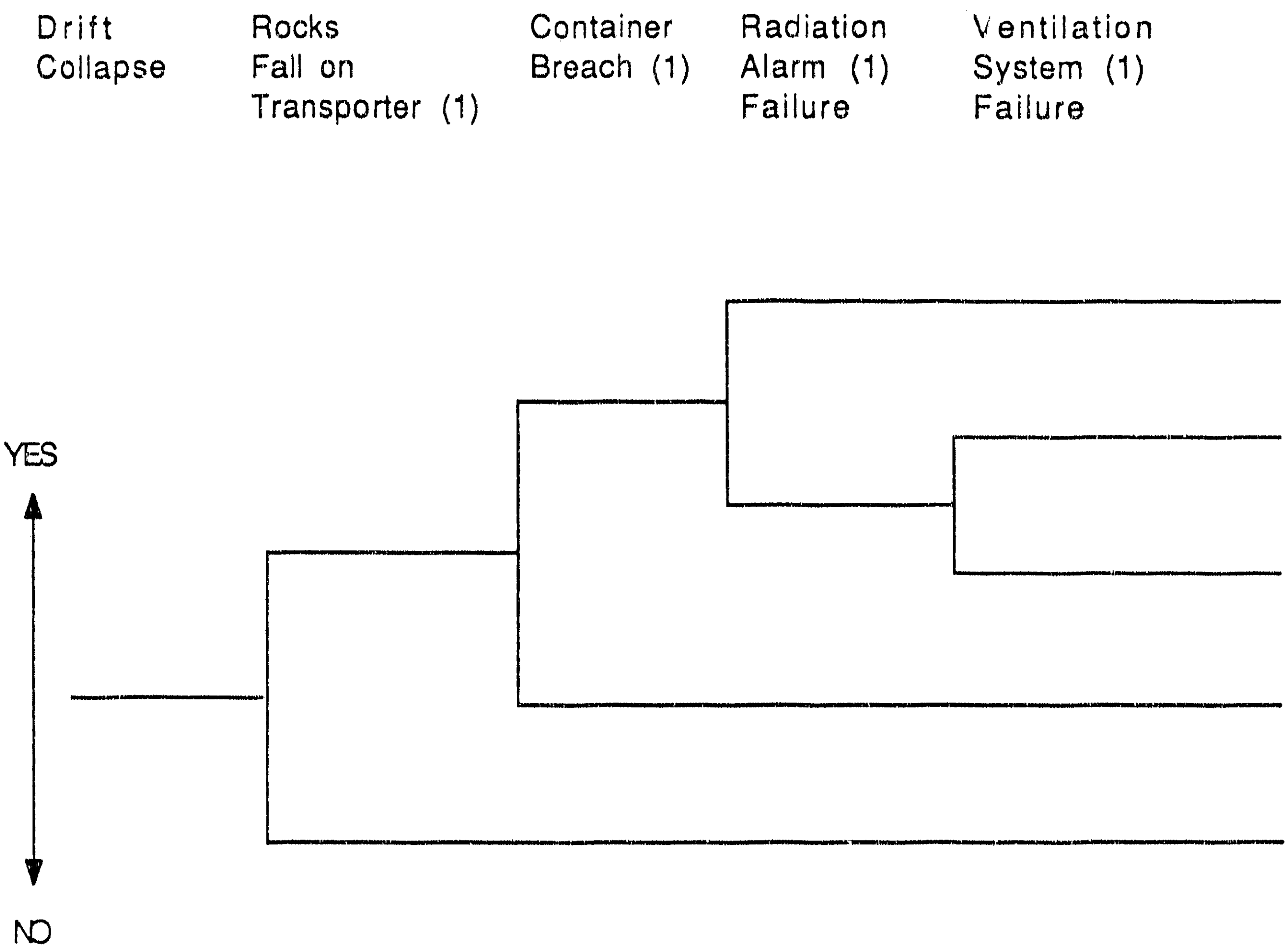
4.3.2 Event Trees

In this subsection, the accident scenarios associated with a drift collapse or shaft liner collapse are developed for Compartments 1, 2, 3, and 4 using event trees (see Step 5 of Table 3-1). The event trees include failures of converted ESF items in combination with failures of other underground repository items. Results from the detailed study of potential underground accidents for the Yucca Mountain repository (Ma, 1988) are used.

4.3.2.1 <u>Event Tree for Compartment 4</u>. An event tree developed in the previous underground accident study for drift collapse is shown in Figure 4-3. The drift collapse either can occur spontaneously (internal event) or can be due to an earthquake (external event). This event tree is also applicable to the drift collapse that may occur in this compartment. A drift collapse onto a loaded transporter during emplacement or removal of a container may cause the container to drop and breach or may cause the transporter to slide, resulting in a shear of the container and fuel rods. Both events may lead to radiological releases. Other intermediate events include failure of the airborne radioactivity monitoring system (including failure to activate the ventilation bypass HEPA filter system) and failure of the ventilation system (defined as the failure to exhaust through HEPA filters or the possible reversal of airflow).

4.3.2.2 <u>Event Tree for Compartments 1, 2, and 3</u>. Based on the discussions given in Subsection 4.3.1, drift collapse in the main test level (Compartment 1) and in the drift to imbricate normal fault zone (Compartment 3), as well as an exploratory shaft (Compartment 2) concrete liner collapse, will not result in any radiological releases. However, these collapses may affect the offsite dose of a concurrent radiological release from other areas of the underground facilities.

Radiological releases in the underground facilities can be initiated either by internal events or by an earthquake. Accident scenarios that result from these two types of initiating events are described by event trees that are discussed in more detail in Ma (1988). These event trees include intermediate events such as radiation alarm failure or ventilation system failure. These failures can either occur spontaneously or can be induced by earthquakes.



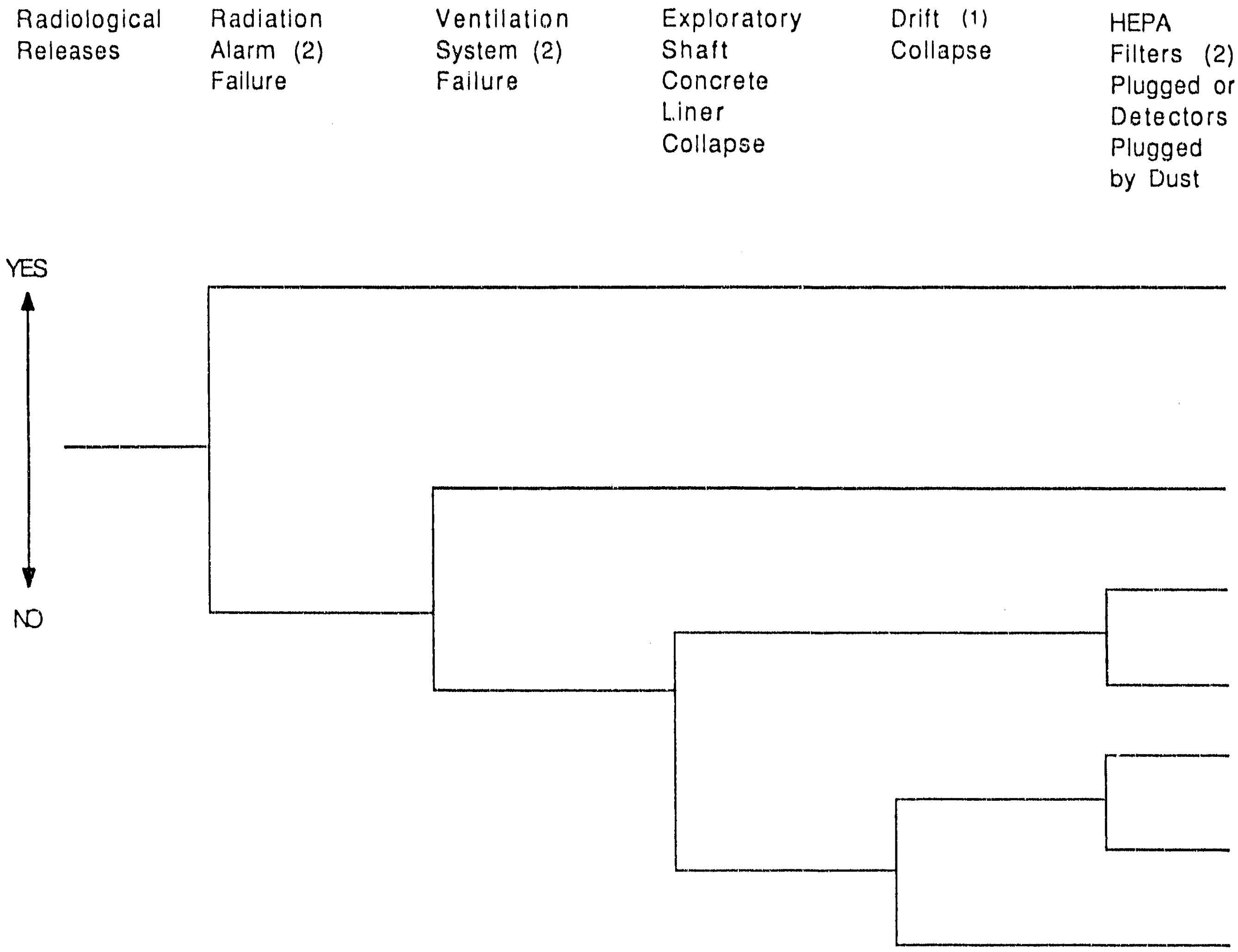
Note: (1) This item is not part of the ESF but is a repository item.

Figure 4-3

Event Tree for Compartment 4

As discussed previously, dust from the collapse of drifts near the exhaust shaft or the exploratory shaft concrete liners may plug the HEPA filters or the filters in the airborne radioactivity monitors. The collapse and the plugging of filters may be treated as one specific failure path leading to the failure of the ventilation system, as in a fault tree analysis, or they may be explicitly incorporated in the event tree, as is done in this report. Figure 4-4 illustrates the accident scenarios either due to an internal event or due to an earthquake that result in radiological releases.

It should be noted that the plugging of either the HEPA filters or the airborne radioactivity monitor filters will result in the leakage or release of radioactive materials to the atmosphere without filtration. Thus, for the purpose of this study, both events are treated together in the event tree.



Note: (1) This drift collapse applies only to the main test level and to the exploratory drift to imbricate normal fault zone.
(2) This item is not part of the ESF but is a repository item.

Figure 4-4

Event Tree for Compartments 1, 2, and 3

5.0 EVENT TREE ANALYSES

This section evaluates the maximum individual offsite dose resulting from each accident scenario developed in Section 4.0 (Step 6 of Table 3-1). Each accident scenario is then classified as either credible or not credible (Step 7 of Table 3-1). The method used to evaluate the radiation doses and probabilities of the underground accident scenarios follows the approach described in Appendix F of MacDougall (1987) and in Ma (1988).

5.1 Radiation Dose Evaluation

Spent fuel accidents can result in releases of three types of radioactive material that are potentially significant to offsite radiation doses: (1) Kr-85 releases, (2) gap radioactivity releases that include Cs, and (3) releases of respirable-size airborne particles (less than 10-micron diameter) of fractured spent fuel. The radionuclide inventories, release and transport mechanisms, and offsite dose evaluations are summarized in this subsection. Additional details of the calculational methods, assumptions, and derivations can be found in Appendix F of MacDougall (1987) and in Ma (1988).

5.1.1 Radionuclide Inventory

Most waste to be emplaced in the repository will be spent fuel from commercial light-water reactors. The spent fuel will be emplaced either as intact fuel assemblies or as consolidated fuel rods.

Defense high level waste (DHLW) resulting from the defense activities of the U.S. government will also be disposed of in the repository. Each DHLW canister will accommodate about 0.5 MTU (equivalent) of vitrified waste. There will also be some vitrified HLW from the West Valley reprocessing plant.

In this study, only PWR spent fuel is considered for radiological releases, as it is assumed to be a bounding case.

According to AP-6.10Q (Appendix A), the radionuclide inventories of the reference PWR spent fuel used in this study are based on the data presented in ORNL/TM-9591 (Roddy, 1986). These data assume a 33,000 MWd/MTU (megawatt-days per MTU) burnup, an initial loading of 0.46 MTU/assembly, and a 10 yr cooling period. Major radionuclide inventories of the PWR spent fuel used in this study are shown in Table 5-1.

5.1.2 Release Mechanisms

The only radioactive gas of concern in 10-yr-old spent fuel is Kr-85, which has a 10.7 yr half life. Because of radioactive decay, quantities of other radioactive gases in the fuel rods are no longer significant. It is assumed that 30 percent of the Kr-85 inventory in a damaged fuel rod will be released if the cladding is ruptured (NRC, 1972).

The two release mechanisms for Cs considered in this analysis are those described by Lorenz (1980a, b). On the basis of Lorenz's empirical model, a breach of spent fuel cladding in a severe thermal environment (about 900°C) will result in a release of 0.028 percent of the total Cs inventory in the fuel rod (burst release). In addition, if the breached fuel is continuously subjected to high temperature, such as in a fire, diffusion of Cs from the main fuel matrix will result in an additional significant release (diffusion release). Lorenz's empirical model indicates that in a period of 1 hr an additional 0.041 percent of the total Cs inventory will be released as a result of this diffusion mechanism at a high temperature of about 1,200°C.

Table 5-1

MAJOR RADIONUCLIDES THAT CONTRIBUTE TO OFFSITE DOSES[a]

Spent Fuel (Ci/Assembly)[b]

Radioisotope	Inventory (Ci/PWR Assembly)[b]
H-3	2.141×10^{2}
C-14	7.152×10^{-1}
Co-60	9.782×10^{2}
Ni-63	3.008×10^{2}
Kr-85	2.238×10^{3}
Sr-90	2.639×10^{4}
Y-90	2.639×10^{4}
I-129	1.453×10^{-2}
Cs-134	2.409×10^{3}
Cs-137	3.788×10^{4}
Ce-144	6.875×10^{1}
Pm-147	4.374×10^{3}
Pu-238	1.075×10^{3}
Pu-239	1.444×10^{2}
Pu-240	2.432×10^{2}
Pu-241	3.580×10^{4}
Am-241	7.798×10^{2}
Cm-244	6.090×10^{2}
Total	1.133×10^{5}

(a) Represents isotopes that contribute 99 percent of the bone dose

(b) Roddy (1986)

Spent fuel accidents involving large mechanical impacts may cause fuel pellet fracture. The resulting UO_2 fuel particles may be released and become a significant airborne source term. Studies of the fracture mechanism by Jardine (1982) and Mecham (1981, 1983), as summarized in Appendix F of MacDougall (1987), found that the fraction of simulated HLW glass specimen fractured into respirable particles (i.e., particles with diameter less than 10 microns) is linearly proportional to the impact energy density:

$$PULF = 2 \times 10^{-4} * E/V$$

where PULF = fraction of unirradiated UO_2 specimen fractured into particulate sizes less than 10 microns (dimensionless)

E = impact energy absorbed by the UO_2 specimen (J)

V = volume of UO_2 specimen (cm^3)

This linear relation is assumed to be valid for spent fuel rods containing many irradiated fuel pellets.

In this study, a mechanical impact of 1 J/cm^3, which is approximately the same as a 35 ft drop or a collision speed of 32 mi/hr, is applied to all accidents for the purposes of evaluating fuel pellet fracture. Therefore, the fraction of spent fuel pellets fractured into respirable particles as a result of an accidental impact is estimated to be 2×10^{-4}.

In some cases, only a fraction of the mechanical impact energy will be absorbed by the fuel pellets. The rest is absorbed by the surrounding materials, such as the cask and transporter. An energy partition factor (EPF) was therefore introduced in the PRSA study (Appendix F of MacDougall, 1987) to represent the fraction of impact energy absorbed by the fuel pellet(s) under various conditions. In cases where an accidental impact involves a runaway transporter, an EPF of 0.2 is assumed. In other cases of this assessment, an EPF of 1.0 is used (i.e., all impact energy is absorbed by the fuel).

Although airborne fuel particles may be generated by an impact, some fraction will be retained by the combination of fuel rod cladding, emplacement container, and transporter cask barriers. Both Appendix F of MacDougall (1987) and this study assume that 10 percent of the airborne fuel particles will escape from a fuel rod with breached cladding (escape factor = 0.1). The escape factors for a breached emplacement container and a transporter cask are also assumed to be 0.1 each.

5.1.3 Transport Mechanisms

The mechanisms for radionuclide transport through the atmospheric pathway are described in this subsection. Depletion of radionuclides inside the underground facilities due to various mechanisms (such as deposition along tunnels and shafts) is neglected in this study. As a result, estimates of radionuclide releases are extremely conservative. A detailed description of transport mechanisms can be found in Appendix F of MacDougall (1987).

When monitors detect radioactivity in the ventilation air, standby HEPA filtration systems at the top of the emplacement area exhaust shaft are automatically activated. The mining and development area is not equipped with any filtration system.

The emplacement area filtration system consists of two stages of HEPA filters in series. The filtration efficiency of one HEPA filter for airborne particles is conservatively taken as 99 percent; the combined filtration efficiency for two HEPA filters in series is taken to be 99.99 percent (Appendix F of MacDougall, 1987). This filtration factor (1×10^{-4}) is applied to all particles (including Cs) released from the emplacement area when HEPA filtration systems are in operation. For Kr-85 releases, a filtration factor of 1 is used because Kr-85 is a noble gas, which is not removed by HEPA filters.

Radionuclides that are released into the atmosphere will be diluted by atmospheric dispersion as they are transported to the site boundary. An atmospheric dispersion factor is used to estimate concentrations of airborne radioactivity downwind from the release point. Appendix F of MacDougall (1987) calculated an atmospheric dispersion factor (X/Q) of 6.4×10^{-5} sec/m^3 for a ground-level release, a 5 km site boundary distance (assumed to be the same as the distance to the nearest unrestricted area boundary), a Pasquill F stability condition, and a wind speed of 1 m/sec, which is based on NRC Regulatory Guide 1.25 (NRC, 1972). This study uses the same atmospheric dispersion factor.

The deposition of radioactive particles during transport to the site boundary causes a reduction in the concentration of airborne particles that reach the site boundary. Appendix F of MacDougall (1987) presented a model which determined that 5 percent of the airborne fuel particles will reach the repository site boundary. This value of 5 percent is also used in this study. It should be noted that no credit is taken for dry deposition in the event that the filtration system is in operation. This is because most of the airborne particles with diameter larger than 1 micron will be removed by the HEPA filters. As a result, the airborne particles released into the atmosphere through the filtration system (mostly particles less than 1 micron in diameter) will have a much smaller deposition factor for a 5 km site boundary distance. NRC Regulatory Guides (1972, 1974b, 1974c, 1977b, 1979a and 1979b) state that, for accident releases, no correction should be made for depletion of radioactive iodine from effluent plumes due to deposition on the ground. The present study considers only the dry deposition for the heavy density UO_2 fuel particles with sizes larger than 1 micron. Dry deposition of UO_2 fuel particles is an important subject for future study.

5.1.4 Offsite Dose Evaluation

The method of estimating the dose to an offsite individual is described in Appendix F of MacDougall (1987), and the same approach is used in this study.

If airborne radioactivity is accidentally released from the repository, an individual could be exposed to radioactivity primarily in two ways: internal exposure from the inhalation of the radioactive material in the plume as it passes the individual; and external exposure from immersion in the plume. Calculations indicate that the immersion doses are insignificant compared to the inhalation doses. Therefore, maximum doses are calculated for the exposed offsite individual due to the inhalation pathway (e.g., assuming the individual is at the nearest site boundary and that this location is directly downwind of the release).

The inhalation dose from a given intake of radioactive material is dependent on the age of the exposed individual. For this initial study, only adult dose conversion factors (DCFs) are used. Although the DCFs for other age groups may be larger, the adult breathing rate is the highest, and adults would receive the maximum doses for those radionuclides that are expected to be released in repository accidents. In addition, all DCFs used in this report reflect 50 yr dose commitments because some inhaled radioactive material may remain in the body for considerable periods following intake. The inhalation DCFs are obtained primarily from Regulatory Guide 1.109 (NRC, 1977a). DCFs not available in Regulatory Guide 1.109 are obtained first from Dunning (1981), then from Holmes (1977), and finally from Killough (1976).

The equation used to calculate the inhalation doses during accidents follows those in NRC Regulatory Guide 1.109 (NRC, 1977a).

$$D_j = BR * \chi/Q * DEP * \sum_i X_i * DCF_{ij}$$

D_j = dose to organ j for an individual due to inhalation (rem)

BR = breathing rate for individuals (m^3/sec)

χ/Q = atmospheric dispersion factor (sec/m^3)

X_i = Curies of isotope i released to the atmosphere as a result of the accident (Ci)

DCF_{ij} = inhalation dose conversion factor for radionuclide i, organ j (rem/Ci inhaled)

DEP = dry deposition factor (dimensionless)

Also, with regard to the offsite dose evaluation, the following assumptions are used:

- o A breathing rate of 3.47×10^{-4} m^3/sec is used in the calculations (NRC, 1972). Based on current NRC guidance, this approach is appropriate when the duration of the release is less than 8 hr.

- o The dry deposition factor applies only to airborne fuel particles and to accident scenarios where the filtration system fails. For airborne fuel particles, DEP = 0.05, for Kr-85 and Cs-137, DEP = 1.

- o The exposure of an offsite individual lasts for the duration of the accidental release, which is assumed to be less than 8 hr.

The radioactivity released to the atmosphere, X_i, can be expressed as follows:

$$X_i = \begin{cases} R_{Kr} * N * A_{Kr} & \text{for krypton} \\ R_{Cs} * N * FIL * A_{Cs} & \text{for cesium} \\ PULF * EPF * ESP * N * FIL * A_i & \text{for particles of fractured fuel} \end{cases}$$

where

A_i = inventory of radionuclide i in a PWR fuel assembly (Ci)

N = number of fuel assemblies breached during the accident for a waste container = 6

R_{Kr} = fraction of krypton released = 0.3

R_{Cs} = fraction of cesium released = 2.8×10^{-4} (for gap burst releases) plus 4.1×10^{-4} (for gap diffusion releases during accidents with fire)

ESP = escape factor (ESP) for particles = $ESP_{clad} * ESP_{cont} * ESP_{cask}$ (each factor is assumed to be 0.1, as appropriate)

FIL = filtration factor for HEPA filters = 10^{-4} (as appropriate)

EPF = energy partition factor = 0.2 for runaway transporter and 1.0 for other impacts.

PULF = fraction of spent fuel fractured into respirable particles = 2×10^{-4}

Radiation doses due to Kr, Cs, and fractured fuel particles are calculated to determine maximum organ doses for various accidents based on the above formulas. The critical organ doses are lung doses due to Kr-85 and bone doses due to both Cs and airborne fuel particles.

5.1.5 Accident Doses at the Unrestricted Area Boundary

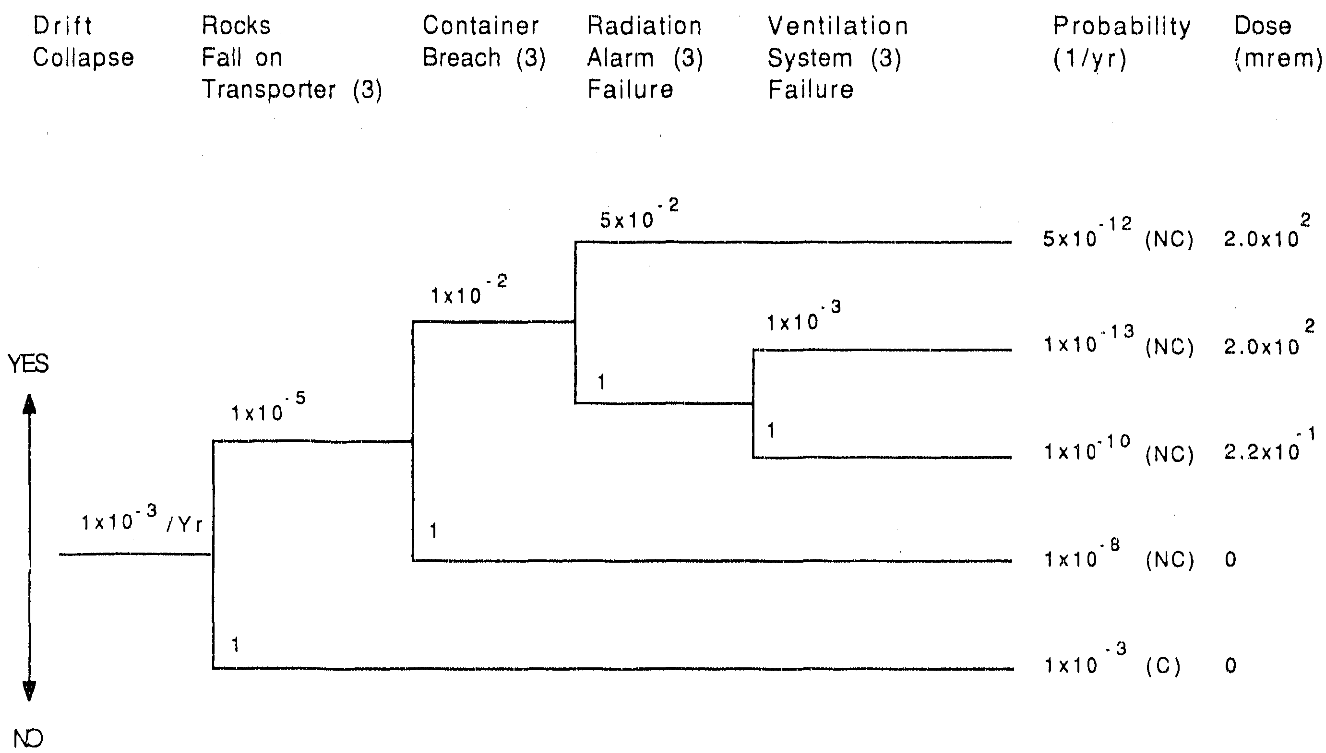
The above dose model was used to calculate the offsite doses for various underground accident scenarios (Ma, 1988). The results are described below.

5.1.5.1 Collapse of the Exploratory Drift to the Ghost Dance Fault. The individual doses at the site boundary due to radiological releases following a drift collapse in the exploratory drift to the Ghost Dance fault are given in Figures 5-1 and 5-2. Figure 5-1 illustrates a spontaneous drift collapse (internal initiating event), while Figure 5-2 represents a drift collapse due to an earthquake (external initiating event). The offsite doses are the same in both cases; however, the annual probabilities are different (see Subsection 5.2).

The results indicate that the dose for scenarios with the ventilation system operating is 0.22 mrem, whereas the dose for scenarios with the ventilation system not operating is 200 mrem. The above values represent doses to the bone, which is the critical organ for accidental releases of Cs and airborne fuel particles.

5.1.5.2 Collapse of the Exploratory Shaft Liner or Drift in Compartments 1, 2, and 3. A collapse of a drift in the main test level, the drift to the imbricate normal fault zone, or the exploratory shaft concrete liner will not result in any radiological releases. However, as discussed in Subsection 4.3, these collapses may generate excessive dust from the underground surfaces and hence may affect the offsite dose due to a concurrent radiological release from other underground areas (Figure 4-4).

Offsite doses due to underground releases for 16 internal initiating events and offsite doses due to 7 types of earthquake-induced releases were evaluated in the underground accident study (Ma, 1988). The largest doses with or without the failure of the ventilation were calculated to be 220 mrem and 0.22 mrem, respectively. These doses at the site boundary are applied to the accident scenarios given in Figures 5-3 and 5-4. Figure 5-3 describes the accident scenarios initiated by internal events, while Figure 5-4 describes the accident scenarios initiated by an earthquake.

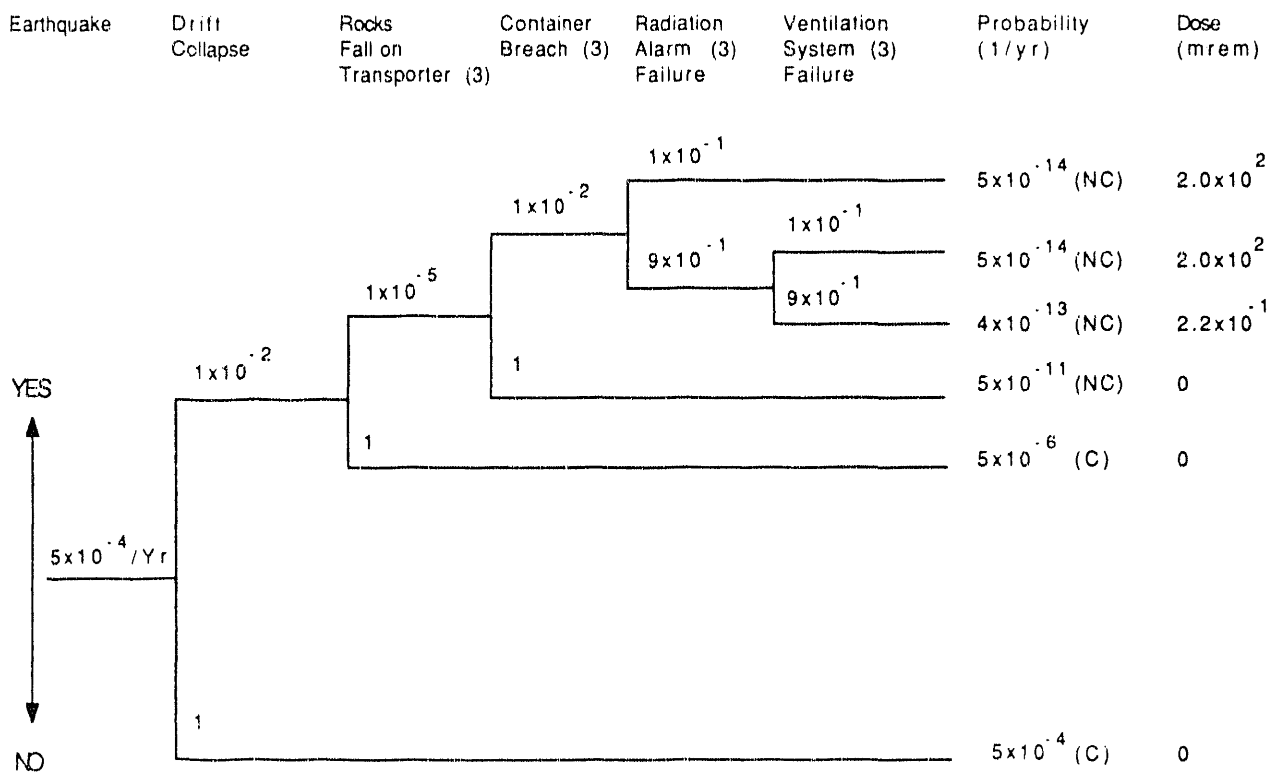


Notes :

(1) All intermediate event probabilities are rounded to one significant figuie; therefore, they may not sum to unity.
(2) NC = Not Credible, C = Credible
(3) This item is not part of the ESF but is a repository item.

Figure 5-1

Event Tree for a Spontaneous Collapse of the Exploratory Drift to the Ghost Dance Fault

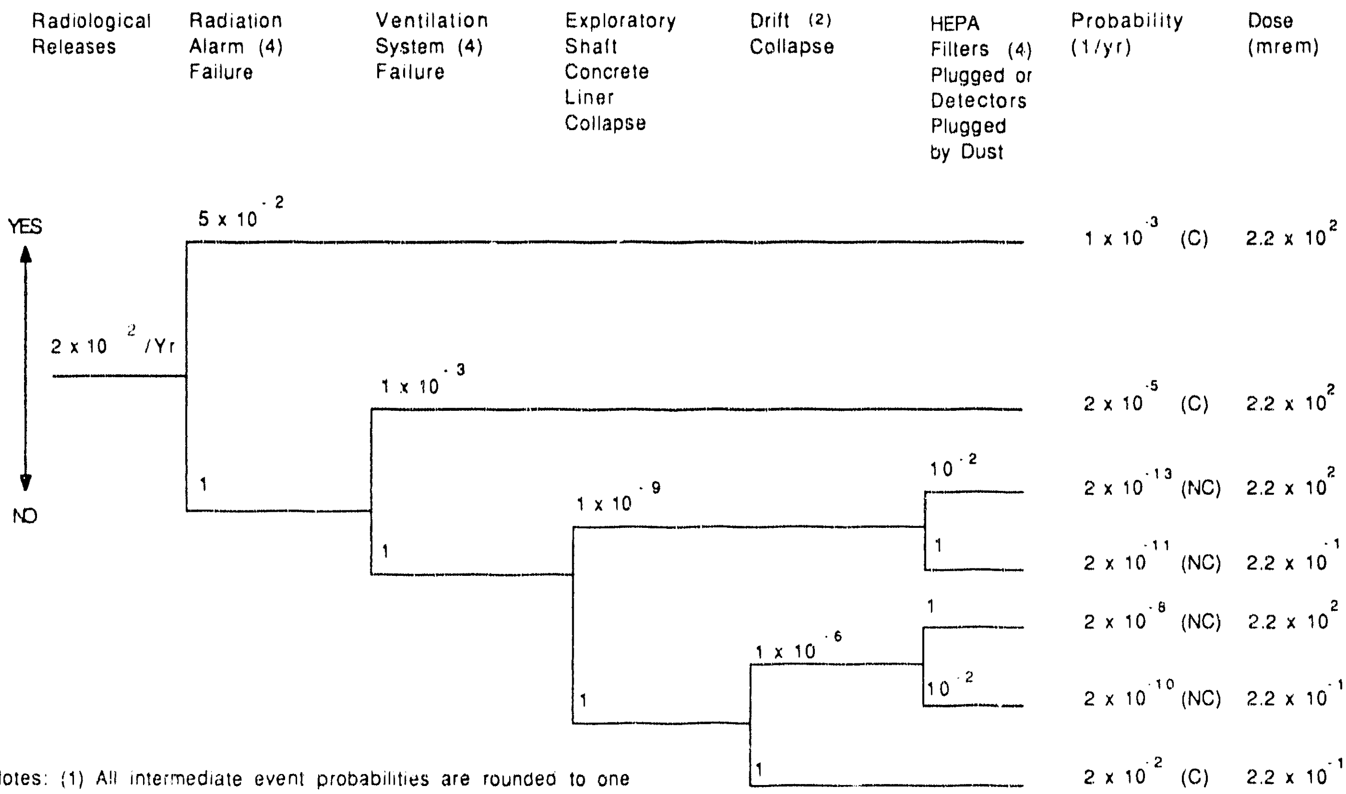


Notes :

(1) All intermediate event probabilities are rounded to one significant figure; therefore, they may not sum to unity.
(2) NC = Not Credible, C = Credible
(3) This item is not part of the ESF but is a repository item.

Figure 5-2

Event Tree for an Earthquake-Induced Collapse of the Exploratory Drift to the Ghost Dance Fault



Notes: (1) All intermediate event probabilities are rounded to one significant figure; therefore, they may not sum to unity.

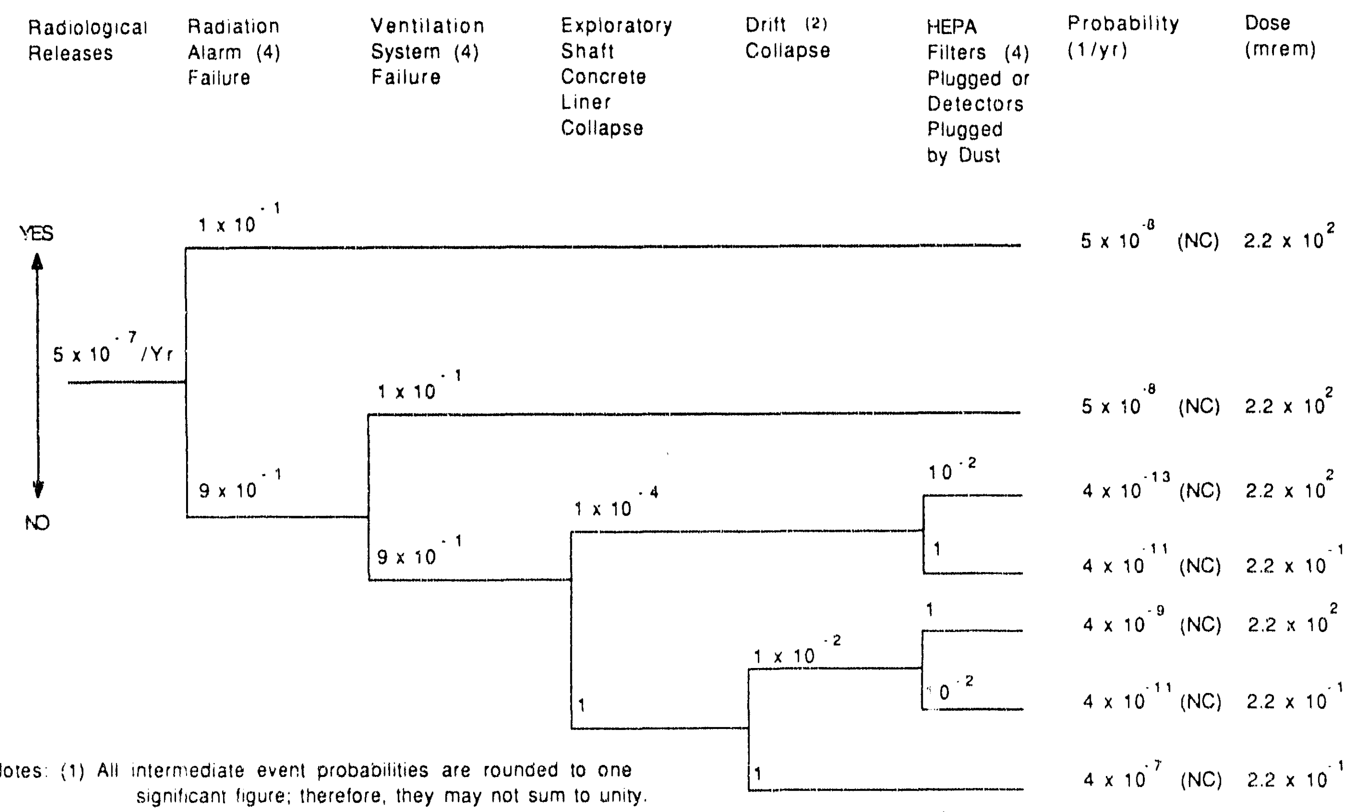
(2) This drift collapse applies only to the main test level and to the exploratory drift to imbricate normal fault zone.

(3) NC = Not Credible, C = Credible

(4) This item is not part of the ESF but is a repository item.

Figure 5-3

Event Tree for Radiological Releases Due to 16 Internal Events



Notes: (1) All intermediate event probabilities are rounded to one significant figure; therefore, they may not sum to unity.

(2) This drift collapse applies only to the main test level and to the exploratory drift to imbricate normal fault zone.

(3) NC = Not Credible, C = Credible

(4) This item is not part of the ESF but is a repository item.

Figure 5-4

Event Tree for 7 Earthquake-Induced Radiological Releases

5.2 Probability Evaluation

As mentioned in Table 3-1, Step 7, accident scenarios are to be classified as credible or not credible. Only credible scenarios that exceed the dose criterion require further assessment for identifying items important to safety. Procedure AP-6.10Q (see Appendix A) states that it is sufficient to denote an event as either credible or not credible, and it is not required (only optional) to determine numerical probabilities for events and scenarios in the event trees. AP-6.10Q also states that a numerical probability of occurrence greater than 1 x 10^{-6}/yr is to be regarded as "credible." Numerical probabilities of various underground accident scenarios were estimated in the underground accident report (Ma, 1988). These estimates were based on accepted predictive techniques, documented judgments of engineers and technical specialists experienced in nuclear facility designs and their potential failure modes, and previously published data of equipment failure. This approach is also adopted in this study. The results are discussed below, and additional details can be found in Ma (1988).

5.2.1 Collapse of the Exploratory Drift to the Ghost Dance Fault

The annual probabilities of accident scenarios initiated by a spontaneous collapse of the exploratory drift to the Ghost Dance fault are depicted in Figure 5-1. It is judged that there may be one to three spontaneous drift collapses throughout the repository during the repository's operating life, which is about 30 yr. Since the total length of drifts in the repository is about 100 mi, the probability of a spontaneous drift collapse is about 10^{-3}/yr-mile. Because the length of the exploratory drift to the Ghost Dance fault is about 1,200 ft, the probability of a drift collapse in this compartment is approximately 10^{-3}/yr.

The probability, P, of rocks falling on a transporter (given a drift collapse) is considered to be

$$P = R * F$$

where R is the ratio of the length of the transporter cask (assumed to be 20 ft) to that of the drift (i.e., 1,200 ft), and F is the fraction of time that a transporter is in the drift. The emplacement drifts are 126 ft apart from each other, meaning that there are about 10 intersections (or 10 boreholes) in the exploratory drift to the Ghost Dance fault (see Figure 2-6). Therefore, during a particular year of repository operations, there will be no more than about 10 emplacement operations in this exploratory drift, each of which takes about 100 min (MacDougall, 1987). On the basis of the above information, it is conservatively estimated that, for this compartment, the probability of rocks falling on a transporter during an emplacement operation, given a drift collapse, is about (20 ft/1,200 ft) (1,000 min/yr) or about 10^{-5}.

The probabilities of other intermediate events are discussed in more detail in Ma (1988).

Figure 5-1 indicates that the annual probabilities of accident scenarios that result in radiological releases in the exploratory drift to the Ghost Dance fault are extremely small, ranging from 5×10^{-12}/yr to 1×10^{-10}/yr. It is therefore concluded that these accident scenarios are not credible.

Figure 5-2 gives the annual probabilities of accident scenarios initiated by an earthquake-induced collapse of the exploratory drift to the Ghost Dance fault. The magnitude of the earthquake is assumed to range from 0.4 g to 0.8 g. Given that such a strong earthquake occurs, the probability of a resulting drift collapse in a 1-mi-long tunnel is judged to be 10^{-2}/event (Ma, 1988). The probabilities of other intermediate events are discussed in more detail in Ma (1988).

Figure 5-2 indicates that accidental radiological releases due to seismically-induced drift collapses have extremely small annual probabilities, ranging from 5×10^{-14}/yr to 4×10^{-13}/yr. It is thus concluded that these accident scenarios are not credible.

5.2.2 Exploratory Shaft Liner Collapse and Drift Collapse in Other Areas

In this subsection, an estimate is made of the probabilities of accident scenarios that involve the simultaneous occurrence of underground releases and various failures in the converted ESF. These failures include collapses of the exploratory shaft concrete liner, drifts in the main test level, and the exploratory drift to the imbricate normal fault zone.

Figure 5-3 illustrates the accident scenarios initiated by internal events. The annual probability of an underground release initiated by any one of the 16 internal events identified in Ma (1988) is 2×10^{-2}/yr, which is obtained by summing the annual probabilities of the releases initiated by each of the 16 internal events. For example, the annual probability of an underground release due to either a transporter collision in the waste ramp or a grapple/hoist failure in the emplacement drift is $(1 \times 10^{-1} \times 10^{-4}/\text{yr}) + (5 \times 10^{-2} \times 10^{-2}/\text{yr})$. The conditional probabilities of airborne radioactivity monitor failure and ventilation system failure are 5×10^{-2}/event and 1×10^{-3}/event, respectively, as given in Ma (1988). The annual probability of a converted exploratory shaft liner collapse is judged to be not credible, or about 10^{-6}/yr.

After an accidental release, the radioactive airborne particles in the underground facility will be carried away by ventilation and reduced to insignificant levels (by about a factor of 10,000) within an estimated time of less than 8 hr. Thus, the total duration of an underground radiological release is taken to be about 8 hr. If a liner collapses within this 8 hr period, the two events can be considered as concurrent events. The probability of a liner collapse within a given 8 hr period is about $(10^{-6}/\text{yr})$ $(8 \text{ hr})/(8{,}760 \text{ hr/yr})$, or 10^{-9}/event.

The total length of the drifts in the main test level and the exploratory drift to the imbricate normal fault zone is about 1 mi. Since it was estimated that the annual probability of a spontaneous drift collapse is about 10^{-3}/yr-mi, the probability of a drift collapse in one of the above two areas is about 10^{-3}/yr. The probability of a drift collapse within a given 8 hr period is therefore about 10^{-6}/event. The probability that HEPA filters will be plugged or that detectors will fail (as a result of dust generated by the collapse of liners or drifts) is assumed to be 10^{-2}/event (unlikely) or 1/event, respectively. The results are given in Figure 5-3.

Figure 5-3 indicates that accident scenarios involving a significant radiological release initiated by any internal event and a concurrent collapse of shaft liner or drift are not credible. The probabilities range from 2×10^{-13}/yr to 2×10^{-8}/yr.

Figure 5-4 illustrates the accident scenarios initiated by an earthquake. Seven types of earthquake-induced underground releases were identified in Ma (1988). The magnitude of the earthquake is assumed to range from 0.4 g to 0.8 g. The annual probabilities of these seven releases were summed, yielding 5×10^{-7}/yr as the annual probability of an underground release due to an earthquake. The conditional probabilities of airborne radioactivity monitor failure and ventilation system failure were both judged to be 0.1/event (Ma, 1988); these values are also adopted in this study. The conditional probability of exploratory shaft concrete liner collapse is considered to be very unlikely, or 10^{-4}/event. Given an earthquake occurs, the probability of a drift collapse in the main test level or in the exploratory drift to the imbricate normal fault zone is judged to be unlikely, or 10^{-2}/event (Ma, 1988). The probability that HEPA filters will be plugged or that detectors will fail (as a result of dust generated by the collapse of liners or drifts) is assumed to be 10^{-2}/event (unlikely) or 1/event, respectively. These probabilities are applied to the accident scenarios described in Figure 5-4.

The event tree in Figure 5-4 indicates that the annual probability that a release due to an earthquake and a concurrent collapse of a shaft liner or drift is very low, ranging from 4 x 10^{-13}/yr to 4 x 10^{-9}/yr. Thus, these concurrent events are judged to be not credible.

5.3 Results

Figures 5-1 through 5-4 show that potential underground accidents will result in doses at the site boundary no greater than 220 mrem. Because the repository is presently in the conceptual design phase, not many data and design details are available. Therefore, the uncertainty in the results of this study and in Ma (1988) has not been quantified. However, these results are considered to be conservative. Future studies are recommended as more data and design details become available to confirm or revise these results.

The event trees in Figures 5-1 though 5-4 also indicate that the annual probability of a drift collapse in the converted ESF (either spontaneously or induced by an earthquake) resulting in a significant radiological release is very low - less than 10^{-11}/yr. Similarly, the probability of a converted ESF structural failure concurrent with a release from another underground area (initiated either by internal events or an earthquake) is also very low - less than 2 x 10^{-8}/yr. Therefore, accidents involving a structural failure of the converted ESF and an underground release (either directly or indirectly) can be considered as not credible.

6.0 IDENTIFICATION OF ITEMS IMPORTANT TO SAFETY

Based on the results of Section 5.0, items important to safety for the converted ESF are identified in this section. Step 8 through Step 13 listed in Table 3-1 are covered.

6.1 Identification of Q-Scenarios

Step 8 of Table 3-1 states that credible accident scenarios exceeding the dose criterion of 0.5 rem should be classified as Q-scenarios.

The results of the offsite dose calculation in Subsection 5.1 indicate that the maximum individual offsite dose from underground accidents for any compartment is 0.22 rem, which is less than 0.5 rem (see Figures 5-1 through 5-4). In addition, accident scenarios involving a failure of the converted ESF and a radiological release are not credible. Therefore, based on the dose criterion and the probability criterion, none of the converted ESF accident scenarios are Q-scenarios.

Step 9 of Table 3-1 indicates that, in addition to the dose criterion and the probability criterion, other criteria, such as probability of occurrence, historical licensing experience, and consensus judgment, will be used to identify Q-scenarios in order to introduce a degree of conservatism into the assessments. No Q-scenario is identified based on these other criteria.

It is therefore concluded that all accident scenarios involving the failure of the converted ESF are classified as NQ-scenarios (i.e., not Q-scenarios). No Q-scenario is identified in this study.

6.2 Elimination of NQ-Scenarios

Step 10 of Table 3-1 indicates that all NQ-scenarios shall be eliminated from further consideration in identifying items important to safety. Based on the above conclusions in Subsection 6.1, all accident scenarios involving the failure of the converted ESF (as described in Figures 5-1 though 5-4) are NQ-scenarios and therefore do not require further assessment.

6.3 Items Important to Safety

Step 11 of Table 3-1 states that the Q-scenarios identified from Steps 8 and 9 shall be assessed further to identify which of the possible items in the facility design are to be classified as important to safety. Since this study concludes that no accident involving a failure of the converted ESF and a radiological release is a Q-scenario, no item in the ESF is identified as important to safety or needs to be placed on the Yucca Mountain repository Q-list.

Step 12 of Table 3-1 states that a summary list of all items classified as important to safety shall be compiled and documented. This study concludes that there are no items on the list of ESF items classified as important to safety, as shown in Table 6-1. Table 6-2 lists the ESF items that are not important to safety (all ESF items are included on this table). It should be noted that most of these ESF items are not important to safety because they will be removed prior to repository operations (as shown in Table 6-2).

6.4 Areas Requiring Further Evaluation

Step 13 of Table 3-1 indicates that the assessment of identifying items important to safety shall be reviewed, revised, and updated in each design stage. This study should be refined early in the repository Advanced Conceptual Design phase as additional design details become available. Potential accidents and failures such as fires, fracture of fuel rods containing many irradiated fuel pellets, dry deposition, and electrical and instrumentation failures should be assessed in more detail at that time. Also, common-mode failures need to be further evaluated in future studies.

Table 6-1

LIST OF ESF ITEMS CLASSIFIED AS IMPORTANT TO SAFETY

	Compartment	Item	Comments
1.	Main test level	None	No Q-scenarios are identified for this compartment
2.	Exploratory shafts	None	No Q-scenarios are identified for this compartment
3.	Drift to imbricate normal fault zone	None	No Q-scenarios are identified for this compartment
4.	Drift to Ghost Dance fault	None	No Q-scenarios are identified for this compartment
5.	Drift to Drill Hole Wash fault	None	No Q-scenarios are identified for this compartment
6.	Upper demonstration breakdown room	None	No Q-scenarios are identified for this compartment

Table 6-2

LIST OF ESF ITEMS NOT IMPORTANT TO SAFETY

ESF Item	Item Number	Compartment Number(s) (see Note b)
ESF Site	1.2.6.1	Note a
Main Pad	1.2.6.1.1	Note a
Auxiliary Pads	1.2.6.1.2	Note a
Access Roads	1.2.6.1.3	Note a
Site Drainage	1.2.6.1.4	Note a
Surface Utilities	1.2.6.2	Note a
Power Systems	1.2.6.2.1	Note a
Water Systems	1.2.6.2.2	Note a
Sewage Systems	1.2.6.2.3	Note a
Communication Systems	1.2.6.2.4	Note a
Mine Wastewater Systems	1.2.6.2.5	Note a
Compressed Air Systems	1.2.6.2.6	Note a
Surface Facilities	1.2.6.3	Note a
Ventilation System	1.2.6.3.1	Note a
Test Support Facility	1.2.6.3.2	Note a
Sites for Temporary Structures	1.2.6.3.3	Note a
Parking Areas	1.2.6.3.4	Note a
Material Storage Facilities	1.2.6.3.5	Note a
Shop	1.2.6.3.6	Note a
Warehouse	1.2.6.3.7	Note a
Temporary Structures	1.2.6.3.8	Note a
Communications/Data Building	1.2.6.3.9	Note a
First Shaft	1.2.6.4	2
Collar	1.2.6.4.1	2
Lining	1.2.6.4.2	2
Stations	1.2.6.4.3	Note a
Furnishings	1.2.6.4.4	Note a
Hoist System	1.2.6.4.5	Note a
Sump	1.2.6.4.6	Note a
Second Shaft	1.2.6.5	2
Collar	1.2.6.5.1	2
Lining	1.2.6.5.2	2
Stations	1.2.6.5.3	Note a
Furnishings	1.2.6.5.4	Note a
Hoist System	1.2.6.5.5	Note a
Sump	1.2.6.5.6	Note a
Underground Excavations	1.2.6.6	
Operations Support Areas	1.2.6.6.1	1
Test Areas	1.2.6.6.2	3,4,5,6

Table 6-2 (Cont'd)

ESF Item	Item Number	Compartment Number(s) (see Note b)
Underground Support Systems	1.2.6.7	Note a
Power Distribution System	1.2.6.7.1	Note a
Communications System	1.2.6.7.2	Note a
Lighting System	1.2.6.7.3	Note a
Ventilation Distribution System	1.2.6.7.4	Note a
Water Distribution System	1.2.6.7.5	Note a
Mine Wastewater Collection	1.2.6.7.6	Note a
Compressed Air Distribution	1.2.6.7.7	Note a
Fire Protection System	1.2.6.7.8	Note a
Muck Handling System	1.2.6.7.9	Note a
Sanitary Facilities	1.2.6.7.10	Note a
Monitoring and Warning Systems	1.2.6.7.11	Note a
Underground Tests	1.2.6.8	Note a
Integrated Data Acquisition System (IDS)	1.2.6.8.1	Note a

Notes: a. All ESF items will be removed prior to repository operations except underground openings (shafts and excavations), shaft liners, and ground support.

b. ESF compartments are as follows:

1. Main test level
2. Exploratory shafts
3. Drift to imbricate normal fault zone
4. Drift to Ghost Dance fault
5. Drift to Drill Hole Wash fault
6. Upper demonstration breakout room

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Appendix A

PROCEDURE AP-6.10Q, IDENTIFICATION OF ITEMS IMPORTANT TO SAFETY

Title

AP-6.10Q IDENTIFICATION OF ITEMS IMPORTANT TO SAFETY

1.0 PURPOSE AND SCOPE

1.1 The purpose of this procedure is to identify the exploratory shaft facility (ESF) and repository structures, systems, and components important to safety (ITS) which are subject to 10 CFR 60, Subpart G Quality Assurance requirements. This procedure specifies the responsibilities and the methods to be used.

1.2 To determine items important to safety, assessments are applied to the appropriate and available repository design configuration including the incorporation of all ESF items. The assessments evaluate potential preclosure accident conditions during the repository waste-receiving, handling, processing, emplacement, caretaking, performance conformation, and decommissioning operations. References are given that contain examples of the application of such assessments to a repository conceptual design (SAND84-2641-F).

1.3 This procedure is iterated or repeated for each completed design phase of a repository or an ESF in order to review, identify, revise, and establish the final list of items important to safety.

2.0 APPLICABILITY

This procedure applies to the Yucca Mountain Project Office, Project participants and their contractors and subcontractors engaged in either the ESF design and construction, repository design and construction or the preclosure performance assessments of the potential repository accident conditions used to establish the repository items important to safety.

3.0 DEFINITIONS

3.1 ACTIVITIES

3.1.1 Activities means deeds, actions, work, or performance of a specific function or task. In the HLW geologic repository program, the 10 CFR Part 60 Subpart G QA program applies to activities affecting the quality of all systems, structures, and components important to safety, and to the design and characterization of barriers important to waste isolation. These activities include: site characterization, facility and equipment construction, facility operation, performance confirmation, permanent closure, and decontamination and dismantling of surface facilities as they relate to items important to safety and barriers important to waste isolation (10 CFR 60.151). In addition, the pertinent requirements of 10 CFR Part 50 Appendix B apply to all activities affecting the quality of structures, systems, and components important to safety and engineered barriers important to waste isolation. These activities include: designing (including such activities as safety analyses, laboratory testing of waste package materials to characterize their performance, and performance assessments), purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, and modifying. These types of activities do not need to be identified as part of the Q-List or Quality Activities List. However, activities related to natural barriers important to waste isolation should be identified and listed on a Quality Activities List. These activities include: performance assessments, site characterization testing, and activities that may impact the waste isolation capability of the natural barrier. For example, site characterization activities such as exploratory shaft construction, borehole drilling, and other activities that could physically or chemically alter properties of the natural barriers in an adverse way. (NUREG-1318)

3.2 CONSEQUENCE ANALYSIS

Consequence analysis is a method by which the consequences of an event are calculated and expressed in some quantitative way, e.g., money loss, deaths, or quantities of radionuclides released to the accessible environment.

3.3 CREDIBLE EVENT OR CREDIBLE ACCIDENT

"Credible event or credible accident" means an event or accident scenario which needs to be considered in the design of the geologic repository (NUREG-1318).

3.4 DESIGN BASIS ACCIDENT

A design basis accident (DBA) is a set of well-defined postulated accidents chosen to establish or measure the adequacy of the safety design of the facility.

3.5 DETERMINISTIC SAFETY ANALYSIS

"Deterministic safety analysis" is a form of safety analysis intended primarily to generate safety design parameters for a facility rather than to measure its safety. Deterministic safety analyses are characterized by (1) evaluation of accident processes and consequences but not of accident likelihood, (2) the use of selected, representative accidents (generally design basis accidents) rather than a comprehensive, complete set of accidents to which the facility might be subject, and (3) the use of pessimistic assumptions and conservatism intended to ensure the presence of margins in the design (but at some cost to the realism of the analysis).

Enhanced margins in the design provide safety margins to account for uncertainties in the assumptions and inputs to the analyses.

3.6 EVENT TREE ANALYSIS

An event tree analysis defines a comprehensive set of accident sequences that encompasses the effects of all realistic and physically possible potential accidents. By definition, an initiating event is the beginning point in the sequence. Hence, a comprehensive list of accident-initiating events must be compiled to ensure that the event trees properly depict all important sequences.

3.7 EXTERNAL EVENTS

External events are those caused by natural phenomena or human activities external to the repository.

3.8 FAULT TREE ANALYSIS

A fault tree analysis examines the various ways in which a system designed to perform a safety function can fail. Each system identified in the event tree as involved in an accident is examined to determine how failures of components within that system could cause the failure of the entire system (NUREG-1318).

3.9 IMPORTANT TO SAFETY

Important to safety, with reference to structures, systems, and components, means those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure (10 CFR 60.2)

3.10 INITIATING EVENT

An initiating event is the starting point of an accident sequence that is generally depicted in an event tree analysis. Initiating events are also used as the starting point in design basis accidents.

3.11 INTERNAL EVENTS

Internal events are those caused by failures or operator activities at the repository.

3.12 INTERACTION MATRIX

Interaction matrix is a systematic way to develop potential initiating events for each of the system compartments in the repository.

3.13 ITEMS IMPORTANT TO SAFETY

Items important to safety are those engineered structures, systems, and components essential to the prevention or mitigation of an accident that could result in a radiation dose to the whole body, or any organ, of 0.5 rem or greater at or beyond the nearest boundary of the unrestricted area at any time until the completion of permanent closure. (NUREG-1318)

3.14 MITIGATIVE SYSTEM

A mitigative system is any system whose design and function actively or passively reduces the severity or consequences of an event once the event has occurred.

3.15 NON-MECHANISTIC FAILURES

Non-mechanistic failures are postulated failures which are not based on previously observed modes or mechanisms but which are assumed to provide conservatism in safety assessments.

3.16 PREVENTIVE SYSTEM

Preventive means to keep from happening or to avert some occurrence from taking place. Hence, a preventive system is one which anticipates some undesirable occurrence or process and counters it in advance of its actual occurrence.

3.17 PROBABILISTIC RISK ASSESSMENT

Probabilistic risk assessment (PRA) (also called "probabilistic safety analysis") is a structured and methodological analytical approach to safety analysis intended primarily to give a realistic picture of the safety profile or risk of the facility.

3.18 Q-LIST

In the geologic repository program, a list of structures, systems, and components important to safety, and engineered barriers important to waste isolation that must be covered under the QA requirements of 10 CFR 60, Subpart G. (NUREG-1318)

3.19 SAFETY ANALYSIS

A safety analysis is a process to systematically identify the hazards of a DOE operation, to describe and analyze the adequacy of the measures taken to eliminate, control, or mitigate identified hazards, and to analyze and evaluate potential accidents and their associated risks.

3.20 SCENARIO

A scenario is an account or sequence of a projected course of action or event.

3.21 UNDERGROUND FACILITY

Underground facility is the underground structure, including openings and backfill materials, but excluding shafts, boreholes, and their seals. (10 CFR 60.2)

3.22 UNRESTRICTED AREA

An unrestricted area is any area to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, and any area used for residential quarters.

4.0 RESPONSIBILITIES

4.1 YUCCA MOUNTAIN PROJECT MANAGER (PM)

The PM assigns a Technical Project Officer (TPO) or a Project Designee to ensure that the provisions of this procedure are implemented. The PM authorizes modification or creation of the list of Items Important to Safety. From time to time, the PM may direct that technical assessment reviews are conducted on the results of this procedure.

4.2 The PM shall assign the responsibility to the cognizant TPO or a Project Designee to implement this procedure and assign personnel to identify items important to safety in the ESF and the repository designs.

4.3 The Yucca Mountain Project Quality Manager and Systems Branch Chief (or their designees) are responsible for review and approval of the lists of items important to safety, items not important to safety, and any reports completed and approved by the TPO as a result of implementing this procedure. The purpose of the review is to provide assurance that the candidate list is consistent with Project Office and participant procedures. The approval does not indicate authentication of the technical data or interpretations contained in the document, nor does the approval relieve the assigned participant of the responsibility for the defense of technical data or interpretations contained therein.

4.3.1 The PM shall issue the results to the Change Control Board (CCB) for construction and baseline control of the Project Q-List.

4.4 TECHNICAL PROJECT OFFICER (TPO)

The TPO shall assign an appropriately qualified participant staff member (PSM) to perform the assessment and to develop the list of items important to safety. The TPO shall ensure that qualified individuals perform any technical reviews of the completed assessments of the items important to safety. After the PSM completes the assessments, the TPO shall, after review, approve and transmit the lists of items important to safety, items not important to safety, and other assessment documentation to the PM.

4.5 YUCCA MOUNTAIN PROJECT PARTICIPANT STAFF MEMBER(S) (PSM)

The PSM shall assemble a group of people from multiple engineering, technical, and scientific disciplines, including personnel who were not a part of the original design team to implement the AP 6.10Q assessments. The group shall be referred to as the Assessment Team.

4.6 The Assessment Team shall carry out the procedure by evaluating the responses, including the offsite doses consequences, of the facility design for credible accident conditions that might affect the facilities performance. The calculated performance predictions shall be compared with the regulatory dose criteria to determine which items from AP-6.9Q should be classified as items important to safety.

4.7 The Assessment Team shall produce a list of the items classified as important to safety (i.e., a major input for the Q-list). The team shall also produce a report that documents the assessments conducted to implement the procedure. A list shall also be prepared of the items classified as not important to safety.

4.8 After completion of the assessment, the PSM shall review and revise any previous list of items important to safety developed in accordance with AP-6.10Q and/or AP-5.4Q. If a previous assessment has assigned a different Quality Level or classification of an item, the PSM shall notify the cognizant TPO or Project Designee that a change request needs to be initiated for the Q-List maintained by the CCB.

4.9 The lists of items important to safety, items not important to safety, and the supporting report documentation shall be submitted by the PSM to the cognizant TPO or the Project Designee for approval and transmittal to the PM.

4.10 CHANGE CONTROL BOARD (CCB)

The CCB shall receive the approved list of items important to safety from the PM and combine this list with any list of items important to waste isolation from AP-6.8Q to compile the Yucca Mountain Q-list. The CCB will, after their approval, baseline the Q-list and maintain the official Project Q-list.

4.11 Exhibit 1 is a flow chart summarizing these responsibilities discussed in 4.1 to 4.10.

5.0 PROCEDURE

5.1 This procedure generates a list of items important to safety. Exhibit 2 summarizes the major steps involved in the procedure.

5.2 As indicated in Step 1 of Exhibit 2, a documented repository and ESF design configuration shall be selected by the assessment team for the application of this procedure. The assessment team shall document the design documents used in their assessments.

5.3 In Step 2, the documented design configuration shall be separated into small zones or areas called facility and system compartments. The compartments shall be named uniquely and shall be selected to facilitate a systematic assessment process.

5.4 In Step 3, all of the items from AP-6.9Q shall be assigned a compartment location and the results documented.

5.5 In step 4, site specific initiating events shall be identified and screened for applicability to all compartments. Initiating events shall be separated into internal and external initiating events. Lists of credible and significant internal and external initiating events requiring further assessment shall be developed on a compartment-by-compartment basis.

5.6 To establish the internal initiating events in 5.5, at least two methods shall be used to generate the list. The methods and the screening criteria shall be documented. The screening process should not reject a credible event that could lead to a significant radiological release yet should reduce the number of events requiring detailed assessments in Step 5.

5.7 Although not mandatory, survey forms and interaction matrices are two methods that have been used in previous repository assessments to identify internal events. The survey forms document accident scenarios for each compartment that are judged by experienced designers to be credible. Interaction matrices identify items in each compartment and use the items as row and column designators. Each row in the matrix is then analyzed column by column to identify possible interactions between items and then potential initiating events and credible accident scenarios are developed and documented.

5.8 To establish the external initiating events in 5.5, a checklist of a wide spectrum of external events shall be used in conjunction with site-specific screening criteria. The checklist, the screening criteria, and the list of credible initiating events requiring further assessment shall be documented.

5.9 In step 5, event trees shall be developed for each internal and each external event in the screened list to depict, logically and systematically, the various accident scenarios. The intermediate events in the event trees shall represent responses of various items in the facility design that occur after the initiating event and hence continue the accident progression into an accident scenario (NUREG/CR-2300).

5.10 In Step 5, fault trees shall not be developed until the advanced conceptual repository design is completed due to the lack of sufficient design details for their development until the advanced conceptual design is completed. Fault trees shall be used to systematically examine the various ways that a system, an item or a major component can fail and result in an initiating event or an intermediate event in an accident scenario.

5.11 In Step 6, offsite dose consequences shall be calculated for each branch in the event tree. The dose consequences shall be calculated for a 50-yr dose commitment to a maximally exposed member of the offsite public at the nearest boundary of the unrestricted area.

5.12 Assessments shall be conducted to calculate source terms and the associated offsite doses. To establish radioactive source terms, the quantities of radioactive materials present, the chemical and physical forms of radioactive materials, the radionuclide content, and the accident conditions shall be considered. Estimates of release fractions of radionuclides for each specific accident scenario shall be made and documented based on their physical and chemical properties and the accident conditions at the time of the release.

5.13 The dose assessments shall be calculated as the total of the external exposure from the passing cloud and the internal exposure from inhalation of radionuclides in the cloud. Dose calculations shall be performed using:

1. X/Q values obtained from Regulatory Guide 1.25 and Regulatory Guide 1.3.

2. Immersion 50-yr dose conversion factors obtained from Regulatory Guide 1.109 and NUREG/CR-1918.

3. Internal 50-yr dose equivalent conversion factors obtained from Regulatory Guide 1.109; NUREG/CR-0150, Volume 3, and NUREG/CR-0172.

4. The radionuclide inventory (Ci/MTU) of the spent fuel shall be obtained from ORNL/TM-9591. If site meteorology is available, the X/Q from Regulatory Guide 1.145 may be used to establish the dose if radioactive plume meander and directionality are to be taken into account.

5.14 In Step 7, the probability or frequency of occurrence of the accident scenarios in the event trees shall be classified. It is sufficient to denote these events as either credible or not credible. It is not required to determine a numerical probability for external, internal, and intermediate events in the event trees. Similarly, numerical values for fault trees are not required.

5.15 Assessments of the probability of occurrences of initiating and intermediate events shall be based on the following considerations:

1. Use of existing or published data.
2. Accepted predictive techniques.
3. Analyses of the performance of the system, and
4. Engineering judgment and experience.

5.16 The probability assessments may utilize previously published data of equipment failures and documented judgments of engineers and technical specialists experienced in nuclear facility designs and their potential failure modes.

5.17 Although not mandatory, standardized forms have been used in previous repository assessments to document judgments.

5.18 The event trees constitute a data base for establishing the list of items important to safety. The regulation 10 CFR 60 provides a single criterion, a dose specification, for identifying items important to safety.

The following two considerations shall be used in Step 8 to identify items important to safety:

1. The dose criterion-an accident scenario must cause an offsite dose of 0.5 rem or greater to merit consideration in identifying items important to safety.

2. The probability criterion-an initiating event (internal or external) or an accident scenario must either be termed "credible" or be estimated to have a probability of occurrence greater than 1×10^{-6}/year to be considered in identifying items important to safety.

5.19 In addition to the two above considerations in 5.18, other considerations shall be used in Step 9 to identify items important to safety based on other project criteria such as:

1. Probability of occurrence.
2. Historical licensing experience.
3. Consensus judgment.

5.20 Using the criteria in 5.18 or 5.19, the event trees shall be assessed in Steps 8, 9, 10, 11, and 12 to identify which items established in the design or AP-6.9Q are important to safety. If the dose screening criterion of 0.5 rem is exceeded in a credible accident scenario, that scenario shall be classified in Step 8 as a Q-scenario. The Q-scenario shall be further assessed in Step 10 to identify specific items important to safety. Scenarios not exceeding these criteria of 5.17 and 5.18 are classified as not Q-scenarios or NQ-scenarios.

5.21 All NQ-scenarios from Step 8 shall be assessed again in Step 9 using the criteria of 5.19 in order to introduce a degree of conservatism into the assessments of items important to safety. Because of this conservatism, which could be unnecessarily excessive, some NQ-scenarios from Step 8 reclassified as Q-scenarios in Step 9 may be reclassified as NQ during a subsequent assessment using this procedure. In such cases, all items involved in the reclassified Q-scenario will be removed from the list of items classified as important to safety.

5.22 For Step 9, scenarios not satisfying the criteria of Step 8 shall be reclassified as Q-scenarios (1) if the scenario is sufficiently similar to others historically classified as Q-scenarios, or (2) when practical considerations based on judgment indicate it could be a Q-scenario, or (3) a calculated probability is sufficiently close to either of the two probability criteria of 5.18 that a variation in assumptions or data could cause either criterion to be exceeded, or (4) when both dose consequences and a calculated probability are sufficiently close to the criteria values in 5.18 that a variation in assumptions or data could cause them to exceed these values. Scenarios not reclassified in Step 9 as Q-scenarios shall remain as NQ-scenarios.

5.23 In Step 10, all NQ-scenarios shall be eliminated from further consideration in identifying items important to safety.

5.24 In Step 11, the Q-scenarios from Steps 8 and 9 shall be assessed further to identify which of the possible items in the facility design or established in AP-6.9Q are to be classified as important to safety. The assessment shall determine which role specific items play in the accident scenarios. These assessments and the rationale for assigning specific items as important to safety shall be documented.

5.25 The assessment in Step 11 shall include a classification of items from AP-6.9Q. The results shall include a summary tabulation of the items compartment location, their classification, and a basis for their classification as either important to safety or not important to safety. Exhibit 3 is a sample format for reporting the summary tabulation of items not important to safety.

5.26 Considerations for classifying specific items as important to safety may include:

1. Their failure directly causes the release of radioactive materials that exceed the 0.5 rem dose criterion.

2. Their failure causes the loss of essential consequence mitigating items that are relied on to lower the probability of exceeding any offsite accident dose limit criterion (e.g., 5 rem) to less than 10^{-6}/year, taking into account the initial failure probability.

5.27 In Step 12, a summary listing of all items classified as important to safety shall be compiled and documented. The sample format for reporting this compilation is shown as Exhibit 4 and shall be referred to as the list of items important to safety.

5.28 The assessment in this procedure is iterative. In the facility design context, iterative means that each stage of design generated in the design description documents shall be assessed using the process in Exhibit 2 and the list of items important to safety (Exhibit 4) revised if necessary.

5.29 In Step 13, any list of items important to safety from an earlier design stage shall be reviewed, revised, and updated to reflect the current design stage and assessment using this procedure. In this iterative design process, some items initially classified as important to safety and hence placed in the Project Q-list will likely be removed and some new items added. This iterative process is illustrated by the feedback loop in Exhibit 2.

5.30 The results from these assessments to identify items as important to safety shall be used to guide the design process by feeding back new requirements to the facility designers or to the design bases (WM-87). Such recommendations from these assessments for new requirements, which should result in an overall improvement in the safety of the repository design, shall be documented and be included in the assessment documentation as recommendations for further evaluation by those responsible for the facility design.

5.31 All source information on which the analyses of items important to safety is based will be listed in the documentation of the results of this procedure and will be baselined as discussed in Section 5.35. This listing must be sufficient to uniquely identify the specific sources of information used.

5.32 To implement this procedure, the PM shall assign the cognizant TPO or the Project Designee to implement this procedure. The TPO shall assign a PSM. The PSM shall appoint an assessment team and conduct the assessments required by this procedure.

5.33 When the PSM completes the assessment, the PSM shall transmit to the TPO for approval the results which include: (1) the list of items important to safety, (2) the list of items not important to safety and (3) any report documentation. The report documentation shall include objective evidence, or reference thereto, demonstrating that each step in the process shown in Exhibit 2 has been completed.

5.34 The TPO shall review, approve, and transmit the results of implementing this procedure to the PM.

5.35 The PM (or assigned Designees) shall, after review, accept the results approved by the TPO. The purpose of the review is to provide assurance that the candidate list is consistent with Project procedures. The approval does not indicate authentication of the technical data or interpretations contained in the document, nor does the approval relieve the assigned participant of the responsibility for the defense of technical data or interpretations contained therein. The PM (or assigned Designees) shall transmit the list of items important to safety and the associated source information (para. 5.31) to the Project Change Control Board to be baselined in accordance with AP-3.3Q. The CCB will transmit the baselined list to Document Control for distribution and control in accordance with AP-1.5Q (Document Control).

5.36 Revisions. During the life of the Project when changes are made in the criteria, data, analyses, etc. that were used to establish the List of Items Important to Safety, the cognizant TPO or his designee, will review these changes and submit a revised candidate list to the PM. The PM will then arrange a review of the revised candidate list and revise the Yucca Mountain Project List of Items Important to Safety as necessary according to applicable Project procedures.

6.0 REFERENCES

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7.0 APPLICABLE FORMS

EXHIBIT 1 - Flowchart of Responsibilities for AP-6.10Q.

EXHIBIT 2 - General Steps: Flow Chart for Identifying Items Important to Safety.

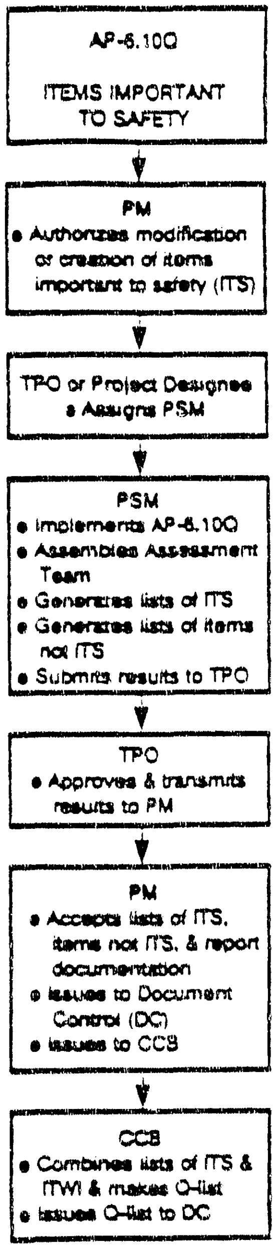
EXHIBIT 3 - Sample Format for List of Items Classified as Not Important to Safety.

EXHIBIT 4 - Sample Format for List of Items Classified as Important to Safety.

8.0 RECORDS

The following documents, used or generated in the implementation of this procedure, have been identified as Quality Assurance Records and shall be forwarded to Central Records Facility for processing in accordance with AP-1.7Q Records Management. Other documents shall be added if identified as QA Records to be maintained by or for DOE.

- o Summary tabulation of items not important to safety
- o List of items important to safety
- o Assessment report documentation that may include:
  - Checklists
  - Screened internal and external events



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ADMINISTRATIVE PROCEDURE

Exhibit 1. Flow Chart of Responsibilities for AP-6.10Q.

Title

AP-6.10Q IDENTIFICATION OF ITEMS IMPORTANT TO SAFETY

Step 1: Select documented design configuration.

Step 2: Define facility and system compartments.

Step 3: Assign compartment locations to items from AP-6.9Q.

Step 4: Identify and screen initiating events to establish credible and significant internal and external events.

Step 5: Develop event trees for accident scenarios. If necessary, develop fault trees.

Step 6: Estimate dose consequences for event trees.

Step 7: Classify accident scenarios as (1) credible, (2) not credible or (3) make (optional) qualitative estimates of frequency of occurrences.

Step 8: Identify credible scenarios in event trees that exceed dose criterion and denote as Q-scenarios requiring further assessment.

Step 9: Identify any other scenarios in event trees that exceed other project criteria and denote as Q-scenarios requiring further assessment.

Step 10: Eliminate all NQ-scenarios in event trees from further assessment.

Step 11: Evaluate all Q-scenarios to identify specific items important to safety.

Step 12: Construct list of items identified as important to safety.

Step 13: Repeat, or iterate, steps 1 to 12 for the various stages of design and review, revise, and update items previously identified as important to safety.

Exhibit 2. General Steps: Flow Chart for Identifying Items Important to Safety.

CLASSIFICATION OF ITEMS: SUMMARY OF ITEMS NOT IMPORTANT TO SAFETY

ITEM	COMPARTMENT LOCATION(S)	COMMENTS

Exhibit 3. Sample Format for List of Items Classified as Not Important to Safety.

LIST OF ITEMS CLASSIFIED AS IMPORTANT TO SAFETY

ITEM	COMPARTMENT LOCATION(S)	COMMENTS

Exhibit 4. Sample Format for List of Items Classified as Important to Safety.

Appendix B

Information from the Reference Information Base Used in this Report

This report contains no information from the Reference Information Base.

Candidate Information for the Reference Information Base

This report contains no candidate information for the Reference Information Base.

Candidate Information for the Site & Engineering Properties Data Base

This report contains no candidate information for the Site and Engineering Properties Data Base.

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